WASHINGTON STATE UNIVERSITY REACTOR
LICENSE NO. R-76
DOCKET NO. 50-27

Response to Request for Additional Information Regarding HEU/LEU Conversion

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RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION WASHINGTON STATE UNIVERSITY MODIFIED TRIGA REACTOR DOCKET NO. 50-27

The purpose of the following questions and answers is to ensure a complete application under 10 CFR 50.64.

The responses to some questions require revisions in the SAR. A revised copy is included with these responses.

1. **General comment. Please confirm consistency between the safety analysis report text and table numbering because it appears that errors may exist in the text. For example, on page 52 Table 19A is referenced. It appears that it should be Table 20 and Table 20A on page 54 should be labeled Table 20. On page 55 it appears that the reference to Table 19A should be Table 21 as should the reference to Table 20 on page 56.**

There have been multiple collaborators involved in the preparation of the Safety Analysis Report for the HEU to LEU Conversion of the Washington State University Reactor, which has led to the noted inconsistencies in table numbering. Although there are multiple collaborators, it is the responsibility of WSU to promulgate the final, correct version of the report. WSU apologizes for the inconsistent numbering of the tables. A revised version of the SAR, with appropriate corrections, is included with this Response to Request for Additional Information. Corrections to the original text and table numbering are listed as follows:

List of Tables

- Page 47. The reference to Table 26 should refer to Table 27.
- Page 48. The reference to Table 26 should refer to Table 27.
- Page 52. The reference to Table 20 should refer to Tables 21 and 22.
- Page 52. The references to Table 19A should refer to Table 20.
- Page 55. The reference to Table 19A should refer to Table 21.
- Page 56. The reference to Table 20 should refer to Table 21.
- Page 58. The references to Table 21 should refer to Table 22.
- Page 59. The references to Table 22 should refer to Table 23.
- Page 60. The reference to Table 23 should refer to Table 24.
- Page 60. The reference to Table 24 should refer to Table 25.
- Page 60. The reference to Table 25 should refer to Table 26.
- Page 63. The reference to Table 26 should refer to Table 27.
- Page 63. The reference to Table 22 should refer to Table 23.
- Page 64. The reference to Table 23 should refer to Table 24.

- Page 64. The reference to Table 27 should refer to Table 28.
- Page 64. The reference to Table 28 should refer to Table 29.
- Page 65. The references to Table 29 should refer to Table 30.
- Page 66. The reference to Table 27 should refer to Table 28.
- Page 73. The reference to Table 29 should refer to Table 30.
- Page 75. The reference to Table 30 should refer to Table 31.
- Page 80. The reference to Table 31 should refer to Table 32.

2. Section 1.1. Why is the transient rod being replaced as part of the conversion?

Response:

The control elements, including the transient rod, are required by WSU Standard Operating Procedure number 5 to be inspected at intervals not to exceed two years. Due to the unique characteristics and uses of the transient rod (e.g. pulsing) it has been practice at WSU to inspect it more frequently than required—generally at once per year, and sometimes twice per year. Removal and inspection of the transient rod (documented with photographs) have shown that it is scratched and should be replaced; however, the transient rod currently functions as designed. The transient rod is being replaced during this reactor conversion project because it is a convenient time to perform the work.

3. Tables 2 and 3. The fuel is referred to as ZrH rather than ZrH_x. Please clarify.

Response:

Tables 2 and 3 should refer to ZrH_x rather than ZrH.

4. Table 2. Please provide engineering diagrams and specifications for the fuel. Please describe the quality assurance process for production of the fuel.

Response:

Engineering Diagrams and Specifications
The engineering diagrams are included as an attachment to this document, referred to as Appendix A: Fuel Engineering Diagrams.

Quality Assurance Process
Close communication between Washington State University, General Atomics, and Idaho National Laboratory is maintained through the safety analysis development and reactor component design process. This communication ensures that the appropriate technical and functional requirements from the reactor safety basis are carried forward through the design and fabrication of the new reactor components. WSU has verified that the drawings and fabrication

specifications documents indicate the dimensions and design parameters that must be met for each item to be in compliance with the reactor safety basis. These documents are used to definitively communicate each component's requirements to WSU personnel, INL procurement personnel, General Atomic personnel, CERCA personnel, and the QA Engineers from each organization. WSU then relies on the well established QA processes at INL and CERCA to ensure that the final product meets the requirements per the drawings and specifications.

INL has a Quality Assurance Program that meets or exceeds the requirements for procuring items and services as established by the DOE. INL has audited CERCA, as required, to ensure that their QA program meets or exceeds the requirements of NQA-1. These requirements are contained in 10 CFR 830 Subpart A, *Quality Assurance Requirements*, DOE Order 414.1C, *Quality Assurance*, and NQA-1-2000, *Quality Assurance Requirements for Nuclear Facility Applications*. These requirements establish the methods that INL, and their suppliers (CERCA in this case), must use to procure material and fabricate items. More specifically, the process requirements relevant to the procurement of safety related items for the WSU are:

- Items and services shall be procured to meet established requirements and perform as specified. [DOE Order 414.1C, Attachment 2, 3.g. (1)] [10 CFR 830.122 (g) (1)]
- Prospective suppliers shall be evaluated and selected on the basis of specified criteria. [DOE Order 414.1C, Attachment 2, 3.g. (2)] [10 CFR 830.122 (g) (2)]
- Process shall be established and implemented to assure that approved suppliers continue to provide acceptable items and services. [DOE Order 414.1C, Attachment 2, 3.g. (3)] [10 CFR 830.122 (g) (3)]

To complete the process, source inspections of the items are performed by qualified INL Inspectors before the fuel elements are released to WSU, after which, receipt inspections of the items are performed by WSU personnel.

WSU maintains electronic and paper copies of source inspection records for each fuel element, and original receipt inspection records for each fuel element.

Inspection Documents
Copies of the source inspection records and receipt inspection records are included with this Response to Request for Additional Information document as Appendices B and C, respectively.

5. Table 2 and Section 4.5.5. Please clarify if the coolant void coefficient of reactivity is positive or negative.

Response:

When some portion of the coolant in the core is replaced by void the core reactivity decreases. Therefore, the void coefficient of reactivity (1% of the coolant in the core is replace by void) should be negative.

The coolant void coefficient should be expressed as a negative quantity in Table 2 on page 6 and in Section 4.5.5 on page 44.

6. **Table 2 and Section 4.5.10. Table 2 has a calculated maximum pulsed reactivity insertion of $2.02 for mixed core 34A while the historical pulsing data has reactivity additions of $2.15. Please explain**

Response:

The data presented in Table 19 from the SAR is for mixed core 34A, and is based upon calculations performed with the BLOOST code. Table 20 from the SAR provides a larger pool of historical pulsing data. Both tables are reproduced below. The peak temperatures for rod position D4NE were also calculated and have been added to the reproduction of Table 20. A complete discussion of the calculation methodology is presented in the response for RAI Question 22. Comparison of the experimentally determined pulsing data and the modeled data demonstrate the model underestimates peak power, total energy release and peak temperatures.

A systematic analysis of historical pulsing data was undertaken by plotting the peak temperature of the IFE's I D6NW and C4NW as a function of reactivity insertion. The plot is illustrated below.

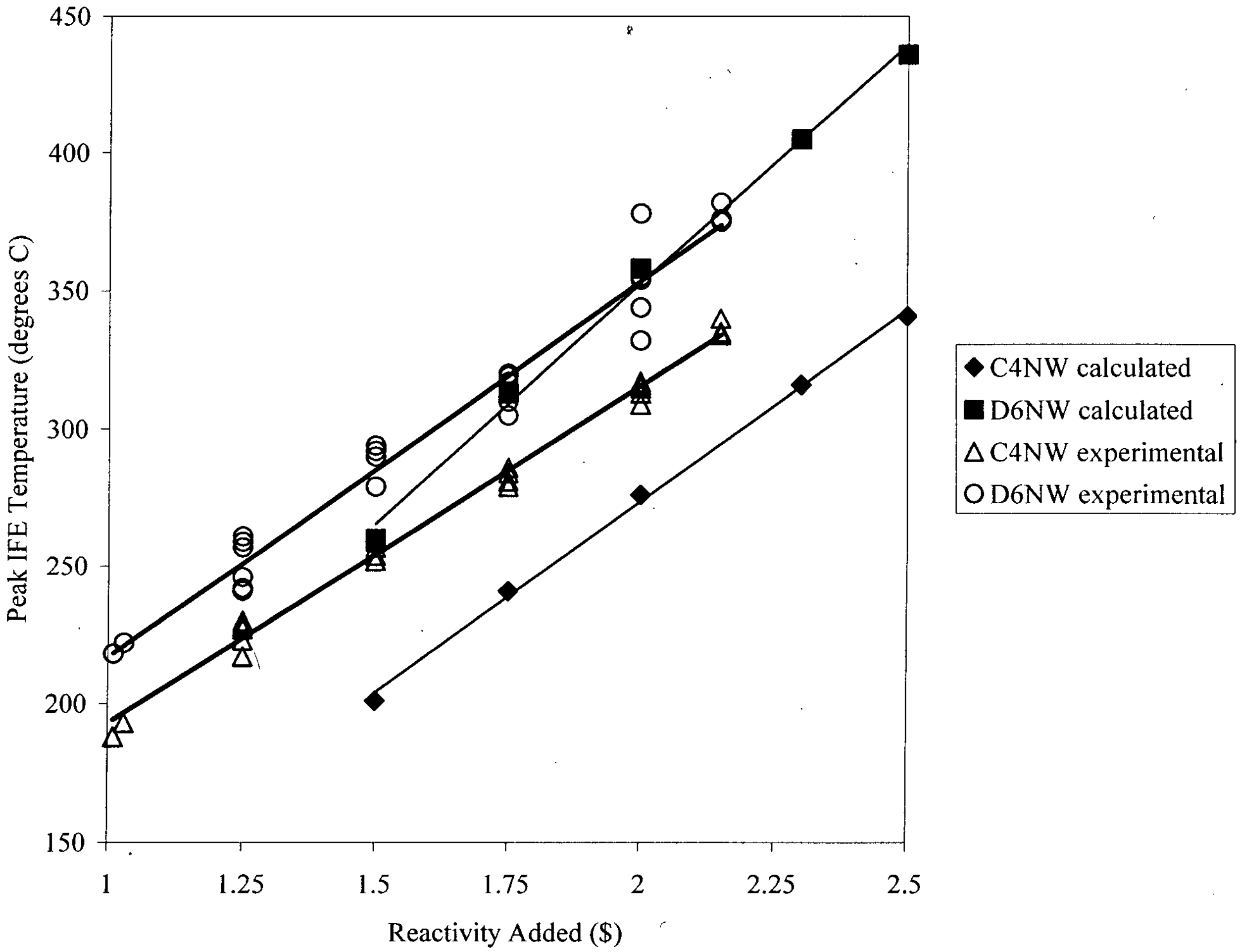
Table 19 Pulse Performance: Measured and Calculated, WSU Mixed HEU Core 34A

Parameter	Pulse				
	$1.50	$1.75	$2.00	$2.30	$ 2.50
Measured Data (a)					
$\hat{P}$ (MW)	240	440	1030		
E(MW – sec)	16	20	25	N/A	N/A
$\hat{T}_{0.3}$ (°C)					
D6NW	279	310	344		
C4NW	254	281	313		
BLOOST-calculation					
$\hat{P}$ (MW)	649	1321	2206	3537	4580
E(MW – sec)	19	26	32	40	45
$\hat{T}$ (°C) (D4NE)	558	701	820	954	1030
$\overline{T}$ core (°C)	201	252	300	356	387
$\hat{T}_{0.3}$ (°C)					
D6NW	260	313	358	405	436
C4NW	201	241	276	316	341

Historical Pulsing Data for WSU Core 34A

Pulse number	Date	Reactivity added	$\hat{T}_{0.3}$ D6NW	$\hat{T}_{0.3}$ C4NW	Peak Power (MW)	Energy (MW•s)	Peak temperature D4NE
1040	11/21/2005	1.25	242	227	60	11.5	414
1041	11/21/2005	2.00	332	317	1200	24	702
1043	11/21/2005	1.50	279	254	-----	16	527
1044	11/21/2005	1.50	279	254	240	16	527
1045	11/21/2005	1.75	310	281	440	20	618
1046	11/21/2005	2.00	344	313	1030	25	722
1047	12/5/2005	1.25	241	217	120	12.4	438
1048	5/31/2006	1.25	246	223	120	11.6	417
1049	5/31/2006	1.75	305	279	500	17.2	555
1050	5/31/2006	2.00	378	309	1000	20	618
1051	11/6/2006	1.25	259	230	160	12	427
1052	11/6/2006	1.50	292	252	260	13	453
1053	11/6/2006	1.75	319	286	480	17.2	555
1054	11/6/2006	2.00	355	317	----	21	640
1055	11/6/2006	2.15	375	334	1420	25	722
1056	11/6/2006	2.15	382	340	1420	24	702
1057	12/14/2006	1.75	317	281	1200	----	----
1058	1/29/2007	1.25	261	228	190	11.9	425
1059	1/29/2007	2.15	376	335	1420	24.8	718
1060	1/29/2007	2.00	355	315	1000	22	661
1061	1/29/2007	0.75	92	78	0	0	----
1062	1/29/2007	1.01	218	188	399	7.8	310
1063	1/29/2007	1.03	222	193	399	8.3	325
1064	5/14/2007	1.25	256	-----	160	11	401
1065	5/14/2007	2.00	354	316	1000	21.5	650
1066	5/14/2007	1.50	292	259	220	14.3	486
1067	5/14/2007	1.75	320	284	440	16.8	546
1068	5/14/2007	1.25	257	229	----	10.5	387
1069	5/14/2007	1.50	290	257	210	14.2	484
1070	8/7/2007	1.50	294	260	300	15	503

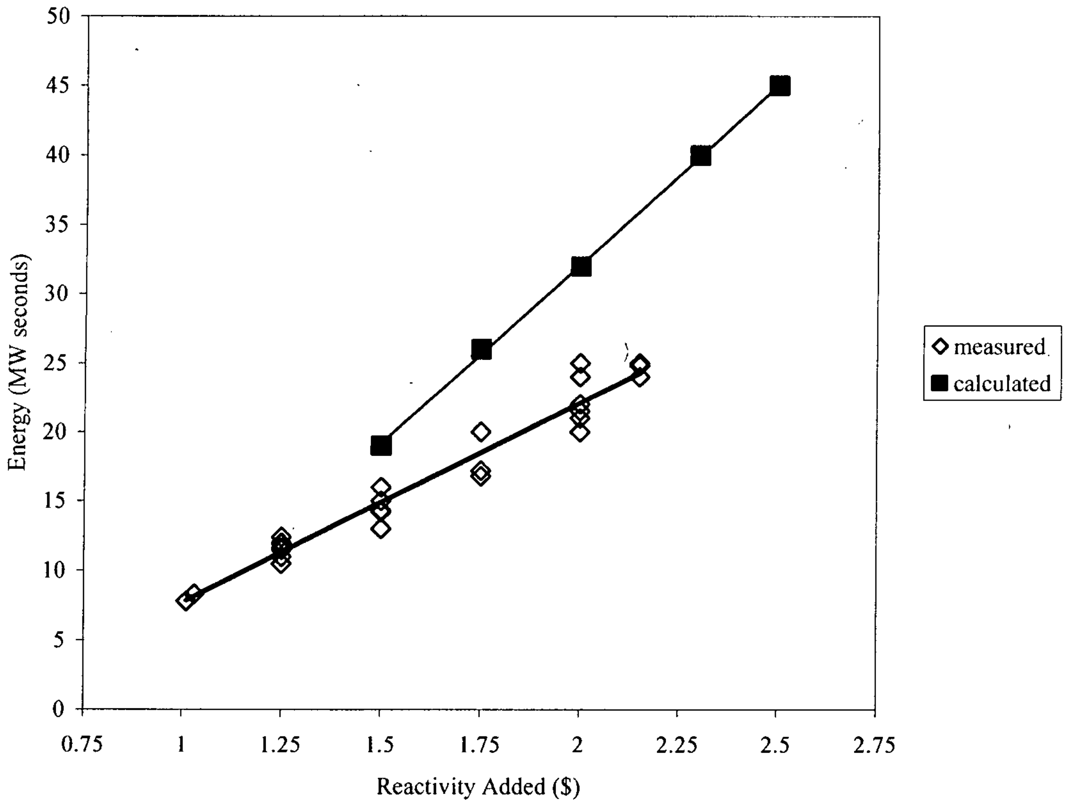
A systematic analysis of historical pulsing data was undertaken by plotting the peak temperature of the IFE's I D6NW and C4NW as a function of reactivity insertion. The calculated and experimentally measured peak temperatures are plotted below.



The equations for the BLOOST calculated best fit lines for the IFE's in core positions D6NW and C4NW are 174° C/$ + 4.4° C and 139° C/$ - 4.9° C, respectively. The equation for the best fit lines for measured peak temperatures for D4NE and C4NW are 136° C/$ + 80° C and 123° C/$ + 70° C, respectively.

Both plots do not give y-intercept values that are close the initial fuel temperature of approximately 30° C. It is apparent that BLOOST under predicts initial fuel temperature, and over predicts the change in fuel temperature as a function of the amount of reactivity added during pulsing, and over predicts peak fuel temperature at pulsing values greater than about \$2. The over prediction of the temperature dependence would be particularly important when using BLOOST to determine upper limits on reactivity insertions for pulsing, and may give a result which is conservative, i.e. over constrains the limitations placed on pulsing due to over prediction of peak fuel temperature values.

BLOOST also considerably underpredicts the peak fuel temperature in C4NW at all reactivity values. This clearly suggests that some fuel positions, such as C4NW are predicted to have lower total energy values, which in turn suggests that other fuel positions must be predicted to have higher energy values. The amount of energy released, as both calculated and measured parameters, are illustrated in the figure below. It is apparent that BLOOST over predicts the energy release by a substantial margin. The experimental data demonstrate that reactivity insertions of \$2.15 for Core 34A liberate slightly less energy than BLOOST calculates for a \$1.75 insertion. For the purposes of testing, Core 35A will be limited to \$2.00 reactivity insertion for pulsing, unless experimental data convincingly demonstrate that greater values may be safely allowed.



7. **Section 4. Is the neutronic analysis done with both graphite/water and water-only reflectors since the Technical Specifications (TSs) allow both designs? Does the presentation in the safety analysis report (SAR) represent the most limiting case? If not, please discuss.**

Response:

The fuel, reflector, and irradiation position arrangement in the grid for Core 34A was chosen to optimize the flux on the west side of the core for irradiations using the thermal column. As a result, graphite reflectors are used on the north and south sides of core 34A, but not on the east or west sides, resulting in graphite and water reflection on the north and south, and water-only reflection on the east and west sides.

The presence of the graphite reflectors increases the reactivity of the core, and therefore represents a limiting case because omission of graphite reflection leads to a core with lower reactivity.

The modeling for cores 34A and 35A was done in a position away from the thermal column, because this location yields the maximum water reflection on the west side, and is therefore the most reactive position. The graphite and water reflection coupled with the choice of core position away from the thermal column yields a water reflector in the north, south, east, and west directions. This yields a model configuration that is the most limiting case because the reactivity is greater when then the reactor is far from the thermal column, versus a position adjacent to the thermal column.

All of the neutronic analyses were done with graphite and water as reflectors, i.e. all references to core configurations 34A and 35A also include graphite reflectors in the grid positions indicated in the respective core grid maps. No analysis was done with water-only configurations. References to reflectors in the SAR for a particular core configuration are always to cores with all the graphite reflectors present. As a result, references to water reflection would also include the presence of graphite when core configuration 34A or 35A is under consideration. References to water reflection in the SAR may be replaced with graphite/water reflection.

8. **Section 4.2.6 and Table 5. Why are the burnup days for the fuel rods calculated? The SAR states that the U-235 in the mass inventory report is used. Please clarify which report this is. In this report, what methodology is used to determine the mass inventory and what is the accuracy of the values used?**

Response:

The burnup days for each fuel rod were calculated to estimate the important fission products in the fuel by using the curve fits shown in Figures 10 thru 15, in order to accurately model the behavior of Core 34A—the current WSU core configuration.

U-235 mass inventory as of September 30, 2005 shown in Table 5 was obtained from WSU.

The calculation for burnup is done on a rod-by-rod basis for each fuel element. The fuel inventory and burnup calculation is done by using operating history from the operating logs and the power distribution, i.e. rod power factors. For example, one megawatt day of burnup is divided among 119 fuel rods, and applying a rod power factor of 1.64 for D6NW under hot critical conditions would give

$$\frac{1\,\mathrm{MWd} \times 1.64}{119\,\mathrm{rods}} = 0.013782\,\mathrm{MWd}$$

A similar calculation is done for each fuel rod to obtain burnup values for each rod. The reports are all kept on file, which provides a burnup history for each fuel rod.

9. **Table 9. The individual calculated rod worths differ from the measured values by 28% for Blade 4 and 111 % for Blade 5. Please explain these differences? What are the estimated (uncertainties) for the measurements of the rod worths?**

Response:

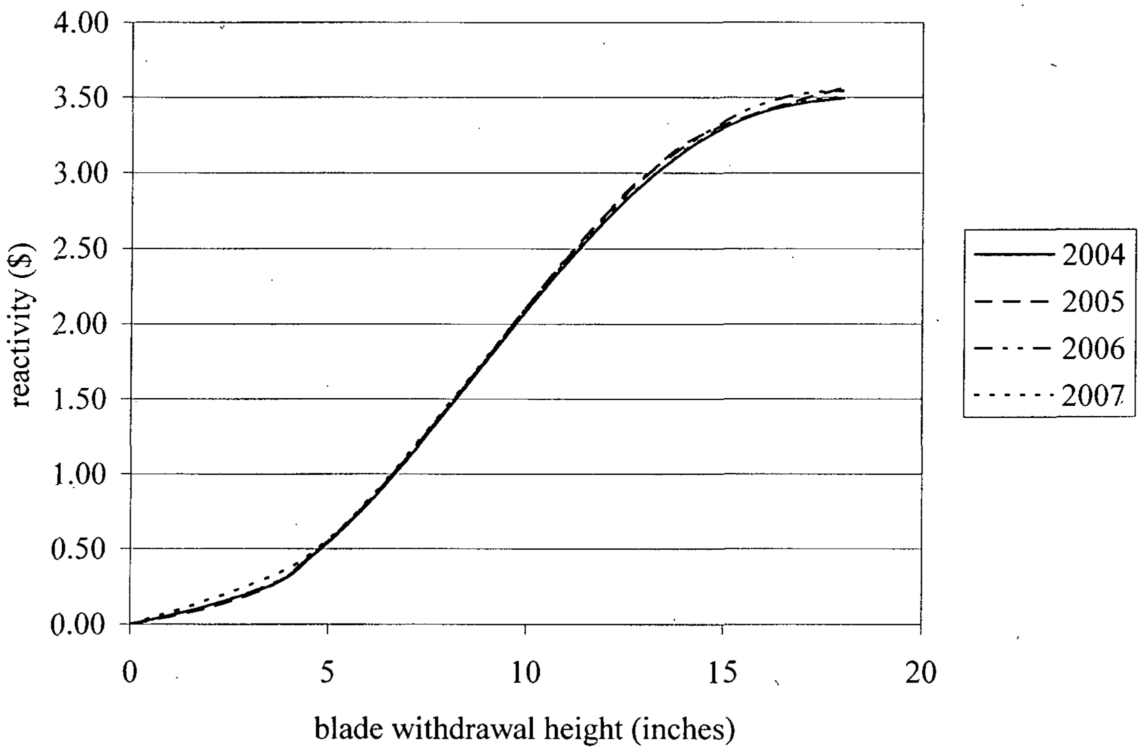
The large difference between the calculated and measured values for blade 5 arises from the fact that blade 5 has a very small worth compared to the other control elements. The absolute value of the blade 5 worth is simply too small to measure accurately; if the measurement error on blade 5 is $0.10 the relative error is about 53%.

The rod worth calculations that were performed for the SAR were carried out by modeling all five control elements in the out position, i.e. completely withdrawn from the core. Additional rod worth calculations were performed in response to this Request for Additional Information. The more recent calculations were conducted with a different control element configuration—namely by assigning each of the control blades and rod in positions wherein the reactor would be approximately critical. The core 34A became almost critical when all blades/rod were inserted 51%.

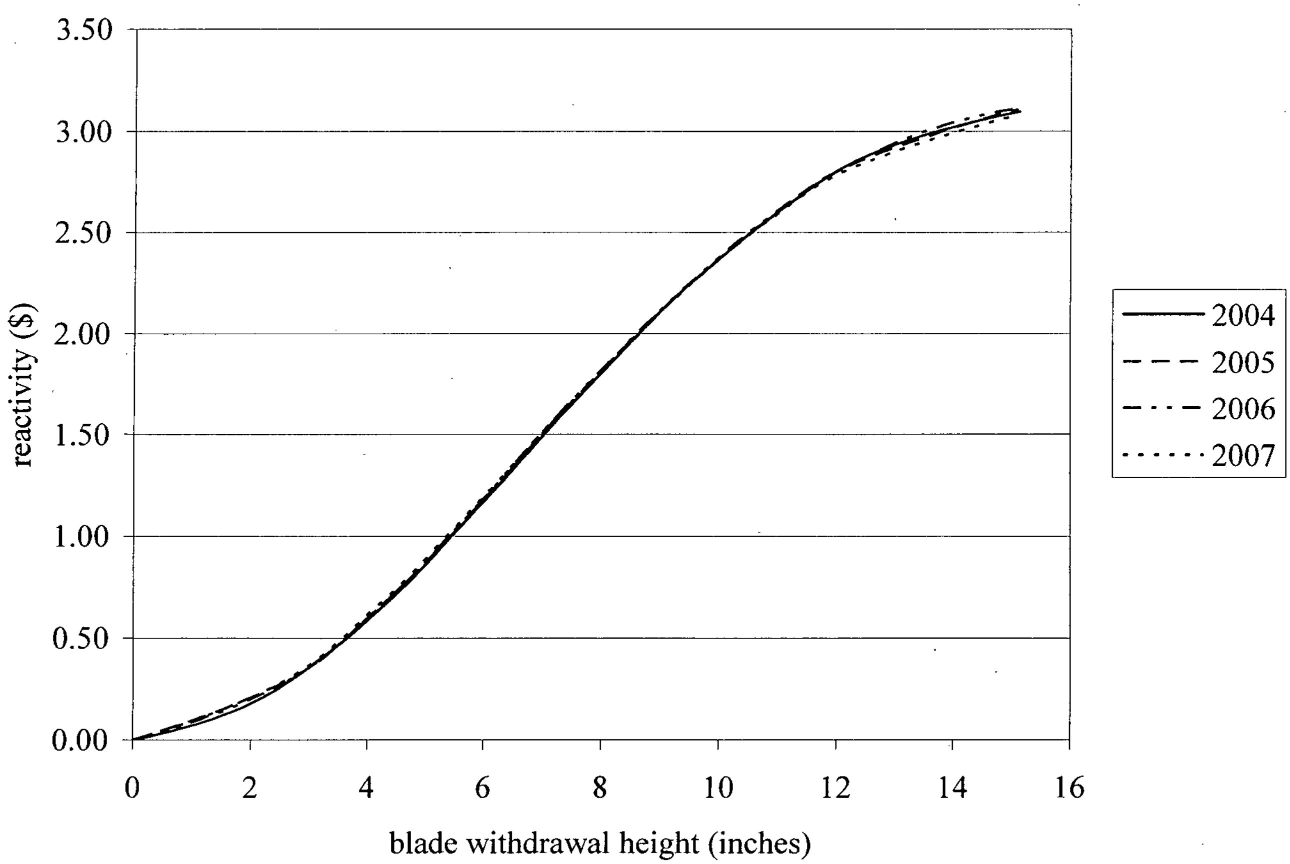
Two MCNP calculations were performed for each blade/rod to calculate the individual blade/rod worths in the new approach; one calculation with the control element fully inserted and one calculation with the control element fully withdrawn. This approach yielded calculated rod worths which closely approximate the experimentally measured values.

WSU has conducted test pulses with reactivity insertion values very close to $1.00, for example $0.75, $1.01, and $1.03 on January 29, 2007. The reactor produced indications of prompt supercriticality (flash of Cerenkov radiation and rapid power rise) for the $1.01 and $1.03 pulses but not for the $0.75 pulse. Control blade and rod calibrations are done by the power doubling time method. Comparison of historical calibration data for the years 2004, 2005, 2006 and 2007 was done to get an estimate of the reproducibility of the calibrations, and thus some estimate of precision of control element worth measurements. The integral rod worth curves for each of the five control elements for the four consecutive years are presented in the following figures.

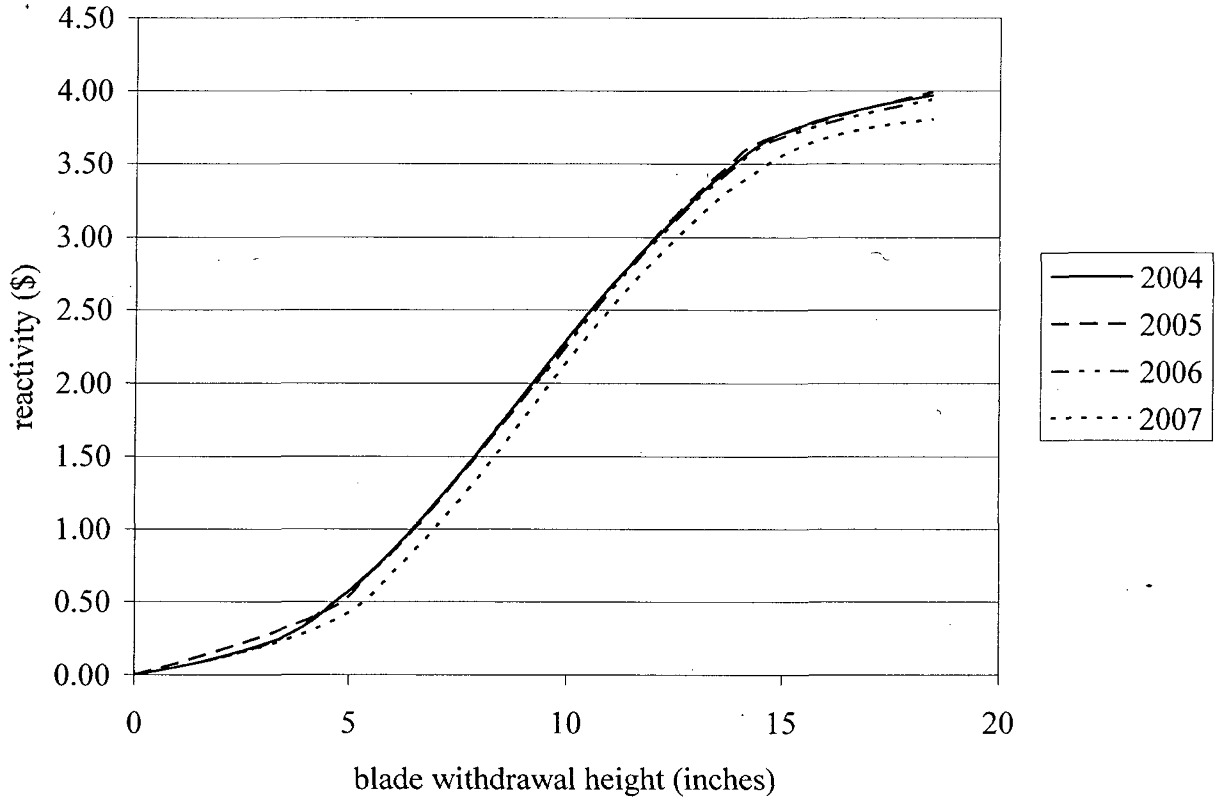
Blade 1 Calibrations for the years 2004, 2005, 2006 and 2007.



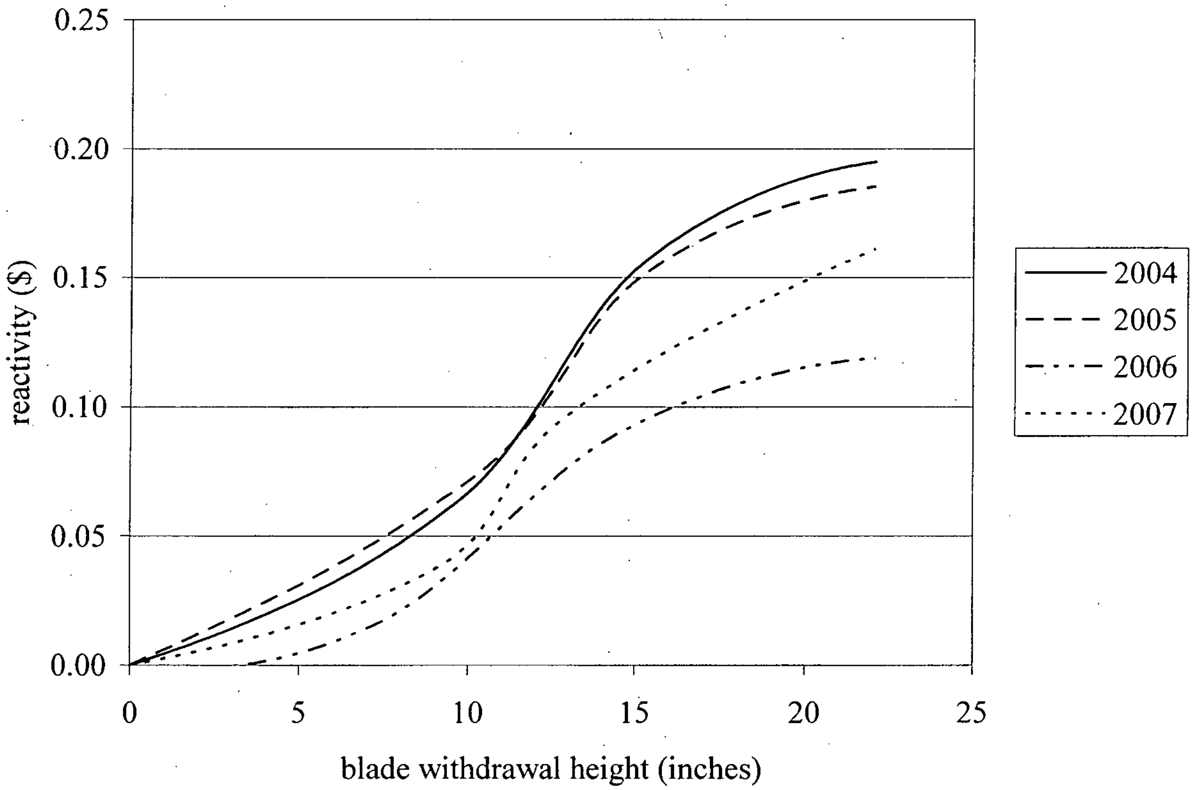
Blade2 Calibrations for the years 2004, 2005, 2006 and 2007.



Rod 3 Calibrations for the years 2004, 2005, 2006 and 2007.



Blade 4 Calibrations for the years 2004, 2005, 2006 and 2007.



Blade 5 Calibrations for the years 2004, 2005, 2006 and 2007.

The WSU reactor is usually operated at 1 MW with blades 1, 2, and 4 withdrawn to heights of about 9.5 to 11.5 inches, depending upon the number of hours of operation and concomitant xenon poisoning. The reactor is operated at full power with control rod 3 (the transient rod) fully withdrawn, and with the regulating blade number 5 at about 11 inches withdrawal height. Due to the higher reactivity worths, the values for control elements numbers 2, 3, and 4 can be measured more precisely than the values for blades 1 and 5. The precision of the measurement also tends to be better near the midrange of blade withdrawal height where the differential rod worth values are greatest. Average values and estimated standard deviations for control element worths at half-height (10 inches for blades 1,2, and 4; 8 inches for the transient rod; 11 inches for the regulating blade number 5) and at full withdrawal are given in the following table. The average and estimated standard deviation values were calculated from the data for the four years 2004, 2005, 2006, and 2007.

Control Element Reactivity Worths at Half Withdrawal

Control Element	Withdrawal Height (inches)	Average Integral Reactivity Worth	Percent Relative Standard Deviation
1	9	0.82 ± 0.03	4
2	9	1.77 ± 0.02	0.9
3	7.5	1.66 ± 0.008	0.5
4	9	1.88 ± 0.08	4
5	11	0.07 ± 0.01	19

Control Element Reactivity Worths at Full Withdrawal

Control Element	Withdrawal Height (inches)	Average Integral Reactivity Worth	Percent Relative Standard Deviation
1	18	1.62 ± 0.04	3
2	18	3.52 ± 0.03	0.9
3	15	3.10 ± 0.01	0.5
4	18	3.93 ± 0.08	2
5	22	0.17 ± 0.03	21

The average values are slightly different from those presented in Table 9 of the SAR, as would be expected. Table 9 should appear as below.

Table 9 WSU Mixed HEU Core 34A – Control Rod Worth

	MCNP Calculated	MCNP Recalculated	Measured
Blade 1 (Shim)	$1.32 ± 0.03	$ 1.46 ± 0.03	$ 1.68
Blade 2 (Shim)	$2.89 ± 0.03	$ 3.64 ± 0.03	$ 3.56
Transient Rod 3	$3.22 ± 0.03	$ 3.86 ± 0.03	$ 3.11
Blade 4 (Shim)	$2.86 ± 0.03	$ 3.82 ± 0.03	$ 3.99
Blade 5 (Servo)	$0.40 ± 0.03	$ 0.46 ± 0.03	$ 0.19
Total	$10.68 ± 0.07	$ 13.24 ± 0.07	$ 12.53

10. Section 4.5.1. The calculated reactivity with all of the control rods fully withdrawn is \$6.31 and with them inserted it is -\$6.91. The measured reactivity for the condition with the rods fully withdrawn was quoted as being \$6.65. The text on pages 35 and 36 states that these values are for water reflection only. Is the graphite reflector usually removed, or was only the benchmarking measurement made with the graphite removed? Alternatively should this read the water/graphite reflector instead of water reflector?

Response:

The calculations and measurements were all made with the reactor away from the thermal column; the usual reactor operating position is at seven feet from the reactor to the face of the thermal column. The reactor position away from the thermal column yields the most reactive configuration due to water reflection on all sides. The reactor is operated with the graphite in place, and all calculations and measurements were made with graphite in place. The text on the referenced pages should indicate use of graphite and water as reflector.

11. Section 4.5.1. Please define "STD." Is it the same as "SFE?"

Response:

The acronyms "STD" and "SFE" do not have the same meaning.

STD refers to a standard TRIGA fuel element that is standard in physical configuration and external dimensions, and is classified as LEU fuel, i.e. less than 20% enriched. A standard TRIGA fuel element may have different uranium loadings, e.g. may have either 8.5% or 30% uranium in the uranium/zirconium hydride fuel matrix.

Traditionally, in General Atomics terminology, a standard fuel element (SFE) refers to TRIGA fuel with 8.5% uranium, at 20% enrichment. Due to the higher uranium loading, a TRIGA fuel element with 20% enrichment and 30% uranium would not be referred to as an SFE.

12. It appears as if the HEU core is analyzed assuming operation at constant power. Does this match the actual operating history? If not, what is the effect on key parameters of the constant power assumption?

Response:

No. The relative burnup of WSU fuel is very low, and the effect is negligible.

13. Section 4.5.1. Should the reference to βeff on page 35 be Section 4.5.5 rather than 4.4.5 as stated?

Response:
Yes, the reference should be to Section 4.5.5 on page 39. The report had a typographical error.

14. Section 4.5.3. On page 37 for the LEU core, to be consistent with the wording used for the HEU core, should the combined reactivity worth of all control elements be quoted as (6.37+7.44=) $13.81 and the $10.97 quoted be defined as the sum of individual worths?

Response:

Yes.

Additional calculations of rod worths have been performed. The recalculated rod worths, along with the original values from Table 10 in the SAR, are listed below. The recalculated sum of the individual control blades and rod is $13.65, which is given the amended version of Table 10.

Table 10 WSU Mixed LEU Core 35A – Control Rod Worth

	MCNP Calculated	MCNP Recalculated
Blade 1 (Shim)	$ 1.34± 0.03	$ 1.56 ± 0.03
Blade 2 (Shim)	$ 2.99 ± 0.03	$ 3.79 ± 0.03
Transient Rod 3	$ 3.19 ± 0.03	$ 3.84 ± 0.03
Blade 4 (Shim)	$ 3.02 ± 0.03	$ 4.01 ± 0.03
Blade 5 (Servo)	$ 0.43 ± 0.03	$ 0.44 ± 0.03
Total	$ 10.97 ± 0.07	$ 13.65 ± 0.07

15. Section 4.5.4. It says that calculated shutdown worth is the excess reactivity minus the worth of rods 1, 2, and 3. Shouldn't it be rods 1, 2, and 4?

The WSU Technical Specifications state that the reactor shall not be operated unless the shutdown margin is greater than $0.25 under the following conditions:

- The highest worth non-secured experiment in its most reactive state
- The highest worth control rod and the regulating rod (if not fully scrammable) fully withdrawn and
- The reactor in the cold critical condition without xenon

The recalculated values for control blade/rod for WSU Mixed Core 35A, along with the most recently experimentally determined values for control element worths are:

	MCNP Recalculated	Measured Worths	Date Measured
Blade 1 (Shim)	$ 1.56 ± 0.03	$ 1.60	12/05/2007
Blade 2 (Shim)	$ 3.79 ± 0.03	$ 3.50	12/06/2007
Transient Rod 3	$ 3.84 ± 0.03	$ 3.08	12/17/2007

Blade 4 (Shim)	$ 4.01 ± 0.03	$ 3.81	12/18/2007
Blade 5 (Servo)	$ 0.44 ± 0.03	$ 0.16	01/02/2008
Total	$ 13.64 ± 0.07	$ 12.15	

The highest worth control blade, according to the version of Table 10 in the SAR, is Number 3—the Transient Rod. However, recalculated values, presented in the table above, give Blade 4 as the highest value control element in Core 35A, which is in closer agreement with the experimentally determined value for Core 34A. Thus, the text should specify that the minimum shutdown margin be determined by subtracting the worth of control elements 1, 2, and 3 from the excess reactivity for cores 34A and 35A.

16. Section 4.5.4. Explain why the shutdown margin calculation does not take into account the removal of the experiment with the maximum non-secured reactivity worth.

Response:

A reactivity worth of $1.00 is the maximum that is allowed for a non-secured experiment. Experiments with uranium are limited by the WSU Technical Specifications to 8 mg of fissile material, which would contribute a negligible amount of positive reactivity. Other types of experiments would contribute negative reactivity.

Analysis was done to examine the consequences of accidental pulsing from full power—the description is given in Section 13.5.3 Accidental Pulsing from Full Power. The maximum pulse size is limited to $3.19, which is the full reactivity of the transient rod. There are several conditions which must be satisfied to accomplish pulsing at full power, including either failure or sabotage of the systems that prevent such an act. The consequences of a $3.19 pulse at full power are:

1. The reactor power increases from 1.0 megawatts to a peak pulsed power of 1614 megawatts.
2. The maximum fuel temperature of 997° C is reached immediately after the peak power.
3. The energy release is about 25.2 megajoules in about 1.0 second when the maximum fuel temperature is reached.
4. At peak fuel temperatures below 1150° C, the strength of the fuel clad maintains clad integrity as long as it remains water cooled.

Removal of an unsecured experiment with a reactivity of $1.00 will result in a much lower peak power, energy release, and peak fuel temperature than conducting a $3.19 pulse, due to two factors:

1. Lower reactivity of the experiment
2. The lower rate of the reactivity change due to the withdrawal time required to physically move the experiment.

The second point is germane; the greater withdrawal time for removing an experiment, compared to the more rapid transient rod drive, will cause an activation of the 125% power scrams and/or high fuel temperature scram to take place, thereby governing the upper limit of power, energy release, and fuel temperature.

The shutdown margin calculation that was described in Section 4.5.4 was done without experiments in the irradiation positions. Experiments with negative reactivity would increase the shutdown margin, and the object of the calculations was to determine, as a boundary condition, whether the modeled core would meet Technical Specification requirements for the minimum shutdown margin without the contribution from experiments.

17. Section 4.5.4. Why are the shutdown margins quoted in Table 2 not consistent with the values stated on page 38?

Response:

The shutdown margin values in Table 2 should be changed to:
Mixed HEU Core 34A: -\$2.61
Mixed LEU Core 35A: -\$2.82

18. Section 4.5.5. For the LEU core, using the stated values of kp = 1.04225 and kt = 1.05019, should βeff be 0.0076 ± 0.0002 instead of 0.0075 ± 0.0002 as stated on page 39. Please explain.

Response:

The difference comes from the round up of the significant digits; when 4 significant digits below zero are used the βeff becomes 0.0075, and if 5 significant digits are used the value becomes 0.0076. Please substitute the values on page 39 with:

For Core 34A,

$k_p = 1.04244 \pm 0.00016$ to $k_p = 1.0424 \pm 0.0002$ and
$k_t = 1.05038 \pm 0.00016$ to $k_t = 1.0504 \pm 0.0002$

For Core 35A,

$k_p = 1.04225 \pm 0.00017$ to $k_p = 1.0423 \pm 0.0002$ and
$k_t = 1.05019 \pm 0.00017$ to $k_t = 1.0502 \pm 0.0002$

19. Section 4.5.5. The discussion of the prompt temperature coefficient of reactivity (pages 40-44) appears to confuse reactor and fuel temperature. Is the coefficient obtained using the core average temperature (including water) or the fuel temperature? If the former, please explain why.

Response:

The coefficient is obtained using the core average fuel temperature which can respond rapidly during a reactivity transient. Water and cladding are excluded from the core average fuel temperature.

20. Table 14. The cold excess reactivity is given as $5.299 for the mixed HEU core, but on page 35 the excess reactivity is claimed to be $6.31. For the mixed LEU core the numbers are $4.959 (Table 15) and $6.37 (p. 36), respectively. Please explain.

Response:

The values in Tables 14 and 15 are from the results of DIF3D calculations, and the other values are from MCNP calculations. The DIF3D uses a diffusion method, and the calculated values have a bias to a lower side comparing to the measured and MCNP results. The values in Tables 14 and 15 should be normalized to the MCNP results.

The revised version of the SAR provides the normalized data in Tables 14 and 15. The two tables are reproduced below:

Calculated and Measured Reactivity Loss, WSU Mixed HEU Core 34A

P(MW)	$ (a)	$ (b)	$\Delta\rho$ ($)$_{calc}$ (a)	$\Delta\rho$ ($)$_{calc}$ (b)	$\Delta\rho$ ($)$_{meas}$
0	5.299	6.31			
0.5	4.431	5.28	0.87	1.03	
1.0	3.845	4.58	1.45	1.73	2.20
1.3	3.716	4.42	1.58	1.88	

(a) calculated by DIF3D
(b) DIF3D results normalized to MCNP results

Calculated Reactivity Loss, WSU Mixed LEU Core 35A

P(MW)	$ (a)	$ (b)	$\Delta\rho$ ($) (a)	$\Delta\rho$ ($) (b)
0	4.959	6.37	0	
0.5	3.129	4.019	1.83	2.35
1.0	2.436	3.129	2.52	3.24
1.3	2.194	2.818	2.76	3.55

(a) calculated by DIF3D
(b) DIF3D results normalized to MCNP results

21. **Sections 4.5.9 and 12.6. What controls are in place with respect to loading new fuel into an existing core? Section 12.6 of the SAR discusses core locations with low peaking factors where new fuel should be introduced into the core. Should fuel additions be limited to these locations by TS? Provide a calculation to indicate the change in the peaking factor when fresh fuel is placed in an EOL core at the worst location or, if limited to the locations discussed in Section 12.6, those locations.**

Response:
Tables 17 and 18 provide Power Peaking Factors for Core 35A at Beginning of Life and End of Life. The two tables are reproduced below:

Table 17 Power Peaking Factors – WSU Mixed LEU Core 35A - BOL

Beginning of Life _ Rods at Critical Positions						
	Cold Critical - 23°C			**Hot Critical - 280°C**		
	RPF	APF	Intra-Rod	RPF	APF	Intra-Rod
Hot Rod	2.56	1.27	1.35	2.47	1.29	1.19
Ave Rod	1.00	1.29	1.55	1.00	1.33	1.55
IFE 1 – D6NW	1.73	1.26	0.51	1.64	1.27	0.45
IFE 2 – C4NW	1.43	1.17	0.51	1.56	1.44	0.45

Table 18 Power Peaking Factors – WSU Mixed LEU Core 35A – All rods out, EOL

End of Life – All Rods Out			
	Hot Critical - 280°C		
	RPF	APF	Intra-Rod
Hot Rod	2.33	1.27	1.29
Ave Rod	1.00	1.25	1.52
IFE 1 – D6NW	1.55	1.26	0.49
IFE 2 – C4NW	1.78	1.24	0.49

The most important factor with respect to placement of fuel in Core 35A is the Rod Power Factor (RPF)—the power generation in a fuel rod (element) relative to the core averaged rod power generation. The worst case scenario would be the introduction of fresh fuel into the core grid location D4, which features the hot rod position, i.e. D4NE. The largest value for the RPF is 2.56 for 23° C and 2.47 for 280° C, both at the beginning of life for Core 35A. As the core life progresses the RPF value decreases, corresponding to the shift of neutron flux. As a result, the limiting scenario is given by the RPF value at BOL, not EOL. Likewise, by interpolation, some point between BOL and EOL would be expected to also not exceed the limiting BOL RPF value.

Analysis shows that fresh fuel may be safely introduced to grid position D4 at BOL, which would provide a boundary condition for fresh fuel introduction, as this circumstance presents the most restrictive set of conditions, namely the highest possible value for RPF. Please refer to the three Tables below for a complete tabulation of the RPF values for each of the core positions. It is not necessary to establish controls for placement of fresh fuel into the core at times after BOL if the condition presenting the worst-case scenario, i.e. fresh fuel in D4 at BOL, is acceptable. The power peaking for placement of fresh fuel into the "most reactive" position was 1% less than the highest BOL peaking factor. As a result, fresh fuel can be introduced into the highest peaking location, and Technical Specification limitations are not needed for the position of fresh fuel introduction.

RPFs for the Core 35A Mixed LEU at EOL

	1		2		3		4		5		6		7	
B	0.23	0.27	0.35	0.43	0.48	0.56					0.51	0.43		
	0.29	0.34	0.50	0.62	0.66	0.73					0.64	0.55		
C	0.38	0.48	0.62	0.69	1.75	1.72	1.78	1.78	1.71	1.61	1.46	1.35	0.52	0.48
	0.42	0.54	0.61	0.66	1.65	1.65	1.74	1.83	2.11	1.66	1.41	1.31	0.54	0.51
D	0.43	0.51	1.39	1.54	1.64	1.80	**1.91**	**2.33**	**TR**	2.15	1.55	1.43	0.58	0.54
	0.43	0.51	1.38	1.54	1.64	1.79	**1.90**	**2.00**	2.28	1.82	1.55	1.44	0.59	0.54
E	0.41	0.53	0.59	0.64	1.64	1.64	1.73	1.75	1.71	1.60	1.42	1.33	0.55	0.51
	0.36	0.46	0.60	0.67	1.72	1.69	1.77	1.78	1.71	1.62	1.49	1.41	0.54	0.50
F	0.28	0.34	0.46	0.57	0.63	0.71					0.67	0.58	0.48	0.39
	0.22	0.26	0.32	0.39	0.47	0.57					0.52	0.42	0.34	0.28

RPFs for the Core 35A Mixed LEU at EOL, Fresh 30/20 LEU cluster in D4

	1		2		3		4		5		6		7	
B	0.23	0.27	0.35	0.43	0.48	0.56					0.50	0.42		
	0.29	0.34	0.50	0.62	0.66	0.73					0.64	0.55		
C	0.38	0.47	0.62	0.69	1.75	1.71	1.78	1.78	1.71	1.61	1.46	1.35	0.52	0.48
	0.42	0.54	0.61	0.65	1.65	1.64	1.72	1.80	2.10	1.66	1.41	1.31	0.54	0.51
D	0.43	0.51	1.39	1.54	1.64	1.78	**1.97**	**2.46**	**TR**	2.14	1.55	1.43	0.58	0.54
	0.43	0.51	1.38	1.53	1.63	1.77	**1.97**	**2.07**	2.25	1.82	1.55	1.44	0.59	0.54
E	0.41	0.53	0.59	0.63	1.63	1.63	1.71	1.74	1.71	1.60	1.42	1.33	0.55	0.51
	0.36	0.46	0.60	0.66	1.72	1.69	1.77	1.77	1.71	1.62	1.49	1.41	0.54	0.50
F	0.28	0.33	0.46	0.57	0.63	0.71					0.67	0.58	0.48	0.39
	0.22	0.26	0.32	0.39	0.47	0.57					0.52	0.42	0.34	0.28

RPFs for the Core 35A Mixed LEU at BOL

	1		2		3		4		5		6		7	
B	0.23	0.27	0.35	0.42	0.46	0.52					0.49	0.42		
	0.28	0.33	0.45	0.55	0.60	0.65					0.63	0.55		
C	0.40	0.50	0.65	0.70	1.58	1.51	1.56	1.57	1.56	1.53	1.44	1.37	0.55	0.51
	0.47	0.62	0.70	0.73	1.72	1.68	1.76	1.86	2.22	1.73	1.49	1.42	0.62	0.58
D	0.50	0.60	1.55	1.67	1.73	1.88	**1.98**	**2.47**	**TR**	2.30	1.64	1.56	0.67	0.63
	0.50	0.59	1.54	1.66	1.73	1.87	**1.97**	**2.07**	2.41	1.89	1.62	1.54	0.67	0.62
E	0.46	0.61	0.68	0.71	1.70	1.65	1.74	1.76	1.72	1.61	1.43	1.37	0.59	0.55
	0.38	0.48	0.62	0.66	1.54	1.47	1.52	1.52	1.47	1.40	1.28	1.22	0.49	0.45
F	0.28	0.32	0.41	0.50	0.55	0.61					0.56	0.48	0.39	0.31
	0.22	0.25	0.31	0.37	0.45	0.52					0.45	0.38	0.30	0.24

	8.5/20 LEU Cluster
	30/20 LEU Cluster
	Graphite Reflector

22. Section 4.5.10. Please provide a copy of Reference 8 in order for us to understand the modeling in BLOOST. Table 19 shows that BLOOST over predicts the energy release and it under predicts the temperature increase for small reactivity insertions. Please discuss.

Response:

The BLOOST manual is included with this Response to Request for Additional Information.

A copy of Reference 8 (West, G.B., et al., "Kinetic Behavior of TRIGA Reactors" Gulf General Atomic Report, GA-7882, 1967) is included with this Response to Request of Additional Information.

The Fuchs-Nordheim variable heat capacity model has been used at WSU to determine pulsing performance for the WSU TRIGA reactor. This is a point model that yields average values for the entire core. Corrections are made for power peaking effects in the hot rod position, D4NE to determine the limiting values for pulsing. The peaking factors that were calculated by General Atomics for cold critical condition (23° C), and those used in the WSU pulsing calculations are 2.56, 1.27, 1.35 for the Rod Power Factor (RPF), Axial Power Factor (APF), and Intra-Rod Power Factor (IPF), respectively. The peaking factors decrease as temperature increases giving values of 2.47, 1.29, and 1.19 for the RPF, APF and IPF, respectively, at 280° C. Thus, the product of the peaking factors decreases from 4.39 to 3.79 at 280° C, indicating that calculations using the peaking factors for 23° C yield a calculated peak temperature value that is higher than the true value.

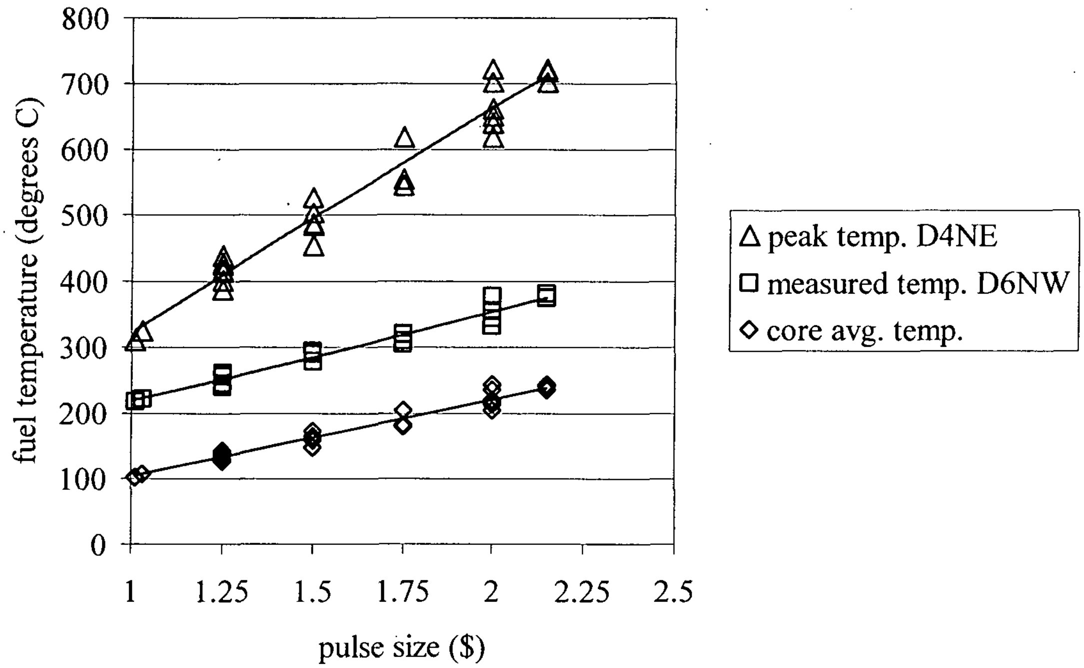
The temperature change of TRIGA fuel for pseudo adiabatic processes such as pulsing is related to the energy release by the Fuchs-Nordheim variable heat capacity model:

$$E_d = 2.08 \times 10^{-3}\,\Delta T^2 + 2.09\Delta T$$

where E_d is the energy density in joules per cubic centimeter of fuel meat and ΔT is the change in average core temperature in degrees Celsius. The relationship between temperature change and energy density can be put into quadratic form and modified to solve for the core average fuel temperature after pulsing, rather than ΔT, for a starting fuel temperature of 25° C

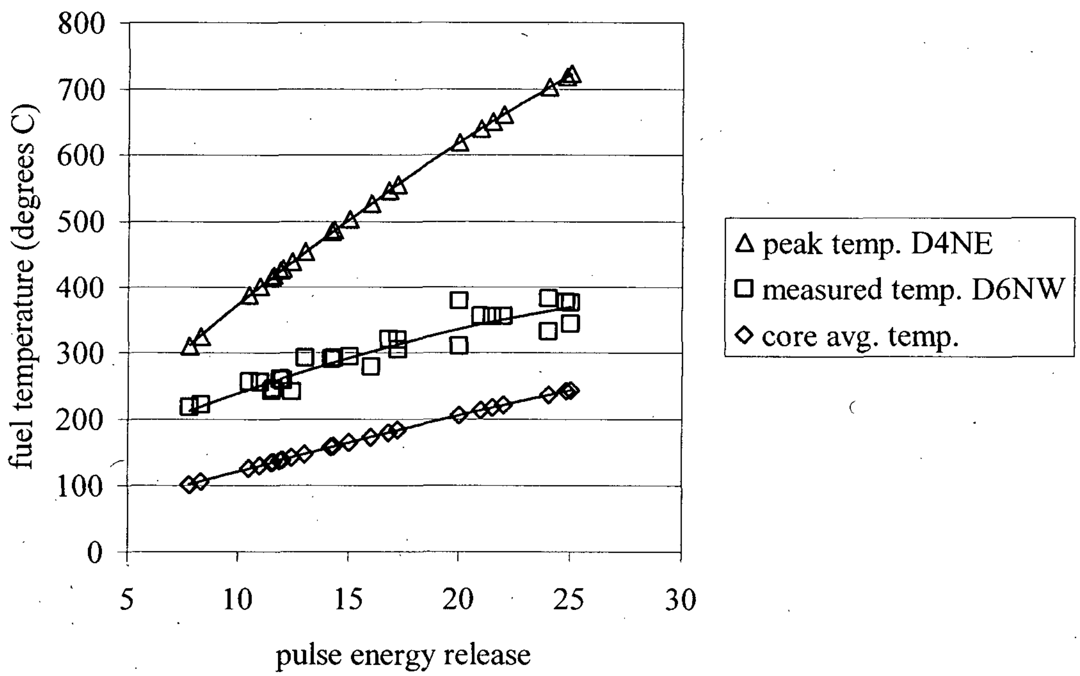
$$2.08 \times 10^{-3}T^2 + 2.09T - 53.6 - E_d = 0$$

Where T is the average fuel temperature after a pulse and the constant term -53.6 corrects the solution of the quadratic equation to an initial fuel temperature of 25° C. The quadratic equation can be used to solve for core average fuel temperature when the energy release for a pulse is known, or can be used to calculate the energy release for a corresponding peak fuel temperature, if the rod power factor, axial power peaking, and intra-rod (or radial) peaking factors are known.



Fuel temperature as a function of pulse size for the WSU TRIGA reactor core 34A.

The dependence of fuel temperature on pulse size for core 34A is illustrated above. The average core temperature was calculated using the Fuchs-Nordheim relationship for measured values of energy release for each pulse. The peaking factors ($2.56 \times 1.27 \times 1.35 = 4.39$ for RPF, APF, IPF) for 23° C were used to calculate the peak energy density in the hottest position in the hot rod, D4NE based on the calculated average core temperature. The peak energy density was used to calculate the peak temperatures in D4NE. The values for D6NW were taken directly from temperature indications on the reactor control console. The peak temperatures for the hottest position in D4NE for pulses of $2.00 and $2.15 are clearly well below the temperature limit of 830° C for pulsing.



Fuel temperatures as functions of pulse energy release.

The pulse energy release was measured and recorded for each pulse. The total energy release was used to calculate an average energy density, which was used in the Fuchs-Nordheim model to calculate the core average temperature. The peaking factors were used to determine the peak energy density in rod D4NE, and the corresponding peak temperatures. The measured temperatures in D6NW were read directly from the temperature indication on the reactor control console. The highest temperature indications are for three \$2.15 pulses, corresponding to total energy releases of 24, 24.8, and 25 megajoules. The peak temperatures for these three pulses are calculated to be 702, 718, and 722 degrees C, respectively. The peak temperatures for the hottest position in D4NE for pulses of \$2.00 and \$2.15 are clearly well below the temperature limit of 830° C for pulsing.

The peak temperatures for each of the pulses are listed in the following Table.

Historical Pulsing Data for WSU Core 34A

Pulse number	Date	Reactivity added	Energy (MW•s)	$\hat{T}_{0.3}$ D6NW	$\hat{T}_{0.3}$ C4NW	$\hat{T}$ D4NE
1040	11/21/2005	1.25	11.5	242	227	414
1041	11/21/2005	2.00	24	332	317	702
1043	11/21/2005	1.50	16	279	254	527
1044	11/21/2005	1.50	16	279	254	527
1045	11/21/2005	1.75	20	310	281	618
1046	11/21/2005	2.00	25	344	313	722
1047	12/5/2005	1.25	12.4	241	217	438
1048	5/31/2006	1.25	11.6	246	223	417
1049	5/31/2006	1.75	17.2	305	279	555
1050	5/31/2006	2.00	20	378	309	618
1051	11/6/2006	1.25	12	259	230	427
1052	11/6/2006	1.50	13	292	252	453
1053	11/6/2006	1.75	17.2	319	286	555
1054	11/6/2006	2.00	21	355	317	640
1055	11/6/2006	2.15	25	375	334	722
1056	11/6/2006	2.15	24	382	340	702
1057	12/14/2006	1.75	----	317	281	----
1058	1/29/2007	1.25	11.9	261	228	425
1059	1/29/2007	2.15	24.8	376	335	718
1060	1/29/2007	2.00	22	355	315	661
1061	1/29/2007	0.75	0	92	78	----
1062	1/29/2007	1.01	7.8	218	188	310
1063	1/29/2007	1.03	8.3	222	193	325
1064	5/14/2007	1.25	11	256	-----	401
1065	5/14/2007	2.00	21.5	354	316	650
1066	5/14/2007	1.50	14.3	292	259	486
1067	5/14/2007	1.75	16.8	320	284	546
1068	5/14/2007	1.25	10.5	257	229	387
1069	5/14/2007	1.50	14.2	290	257	484
1070	8/7/2007	1.50	15	294	260	503

23. Section 4.5.10. Does the analysis of pulsed operation assume that a gap of 0.2 mils will apply to the new core? If so, how would a larger gap affect the pulsing result including peak temperature?

Response:

The analysis for the new core assumes an average gap of 1.75 mils under cold conditions which reduces due to thermal expansion as the fuel temperature increases. The thermal resistance under cold conditions is about 35% greater for the new core (35A) compared to the current core (34A). The effect on pulsing is minor due to the short time scale of the pulse which results in an essentially adiabatic transient during the actual pulse.

24. Section 4.7.2. Provide justification for ignoring cross flow between neighboring flow channels.

Response:

The RELAP5 model of the hot channel simulates the highest power rod in the core and its surrounding coolant. If cross flow between this channel and its neighbors were included in the analysis, then the predicted coolant temperatures in the hot channel would be reduced. Lower coolant temperatures, however, have less buoyancy and cause lower flow rates. Lowering the coolant temperatures increases the critical heat flux while lowering the flow rate moderately reduces the CHF. As discussed below, the reduced temperature is the stronger effect. Therefore, cross flow increases the heat flux at which CHF occurs in the limiting channel.

A pressure difference between a hot and colder channel axially along the channels would provide a cross flow. However, the pressures at the core inlet and core outlet are equal for the two channels. Thus, there is likely little difference in pressure between the two channels as one traverses from the bottom of the core to the top of the core. Hence, a small, if any, cross flow would be expected. A look at the overall buoyancy/friction pressure changes in channels adjacent to the hot channel indicates the cross flow would be from the cold to the hot channel. The hot channel flow rate would increase due to the additional cross flow, however cross flow in this direction would decrease the hot channel density (buoyancy) and diminish the effect of increased hot channel flow rate.

A response to this question on cross flow was provided by Ross Jensen in a RELAP analysis of the McClellan Nuclear Radiation Center (MNRC) reactor for a paper presentation in 1998 under a contract from ANL to support increasing the MNRC operating power. "The RELAP5 code provides a means for estimating the effects of cross flow between the hot and average channels. The cross flow effect is expected to be very small, and it is impossible to asses the accuracy of computed cross flows. Scoping calculations with RELAP5 showed cross flow to have no effect on fuel temperature and <u>to slightly increase</u> the critical heat flux ratio. Thus, cross flow is conservatively neglected in this analysis" (Jensen, 1998).

His conclusions are supported by the recent STAT-RELAP comparison study. That study shows fuel temperatures are only slightly affected by the channel flow rate because the channel is in sub-cooled nuclear boiling. Likewise, there will be some change in the CHFR (DNBR) because there is a velocity (mass flux) effect in the Bernath DNB correlations. Changing the input mass flux by 20% requires a similar change in the wall heat flux (reactor power) by 1.5% to re-achieve a DNBR = 1.0. Increasing the hot channel flow rate due to cross flow would increase the DNBR.

As noted in a Sandia report (Rao, 1994), various experiments have revealed that this cross flow is negligible for tightly packed geometries such as PRNC and TAMU. The references for these experiments are Becker (1969), Silvestri (1966), and Gaspari (1974). The ultimate effect of cross flow mixing is to increase CHFR, so that its neglect through the use of subchannel approach is expected to result in conservative estimate of CHF.

Jensen, R. T., and D. L. Newell, "Thermal Hydraulic Calculations to Support Increase in Operating Power in McClellan Nuclear Radiation Center (MNRC) TRIGA Reactor," 1998 RELAP5 International User's Seminar, College Station, Texas, May 1998.

Rao, D. V., and M. S. El-Genk, "Critical Heat Flux Predictions for the Sandia Annular Core Research Reactor," Sandia report SAND 90-7089, August 1994.

Becker, K. M., G. Hernborn, M. Brodl, and G. Erikson, "Burnout Data for Flow of Boiling Water in Vertical Round Ducts, Annuli and Rod Clusters, AE-177, AB Atomenergi, Sweden, 1969.

Silvestri, M., "On the Burnout Equation and on the Location of Burnout Points," Energia Nucleare, Vol. 13, No. 9, pp. 469-479, 1966.

Gaspari, G. P., A. Hassid, and F. Lucchini, "A Rod Centered Subchannel Analysis with Turbulent Mixing for Critical Heat Flux Prediction in Rod Clusters Cooled by Boiling Water," Paper B6.12, Proc. Fifth Int. Heat Transfer Conf., 1974.

25. Section 4.5.8/4.7.3/4.8.3. Does the peak power density (maximum local heat flux) at steady-state occur in the fuel element (rod) with the maximum rod power?

Response:

Yes. The rod with the maximum Rod Power Factor (RPF) defines the maximum powered rod and also has the maximum peak power density. The other high powered rods have similar axial power factors (APF) and Intra Rod peaking factors (Intra Rod). Thus, the maximum powered rod has the maximum wall heat flux.

26. Section 4.7.3/4.8.3. Are changes in gap properties (e.g., oxidation at the fuel boundary, gas composition) over time taken into account when calculating fuel temperatures? If not, why not?

Response:

Changes in gap properties are not taken into account over time because operational and experimental evidence shows that the highest fuel temperatures occur at the beginning of core life. The most recent evidence is the decline in IFE temperature at TAMU after conversion from FLIP to LEU 30/20 fuel. Over time and depending on fuel temperature and burnup, the gap between the fuel and cladding reduces until there is contact. Experimental data on swelling of $UZrH_x$ fuel is presented in Simnad, 1980 and in the Post-Irradiation Examination (PIE) of TRIGA LEU fuel tested in the Oak Ridge Research Reactor (Simnad, 1986). Oxygen and nitrogen in the gap gas can be consumed by forming either uranium or zirconium compounds. The gap gas composition at the end of irradiation was determined in the TRIGA LEU fuel testing (Simnad, 1986). The time and temperature dependence of the oxygen and nitrogen comsumption is not known.

Simnad, M. T., "The U-ZrHx Alloy: Its Properties and Use in TRIGA Fuel," General Atomics document E-117-833, February, 1980.

Simnad, M. T., and G. B. West, "Post-Irradiation Examination and Evaluation of TRIGA LEU Fuel Irradiated in the Oak Ridge Research Reactor," General Atomics document GA-A18599, May 1986.

27. Section 4.7.3. Please provide a copy of Reference 10 in order for us to understand the modeling in TAC3D.

Response:

The TAC2D manual (Reference 10) is included with this Response to Request for Additional Information.

28. Section 4.8.1. The thermal hydraulic analysis was performed with an assumed water inlet temperature of 30 °C. Based on this analysis, please propose a TS limiting condition on water temperature or explain why a limit on water temperature is not needed.

Response:

Washington State University currently has an administrative limit of 50° C for maximum pool water temperature—the reactor may not be operated with pool water temperatures greater than 50° C. The reactor pool water cooling system has been shown to be capable of indefinitely maintaining the pool water temperature below 50° C when operating at full licensed power, under all ambient weather conditions.

The analysis was repeated for a power level of 1 MW, to show that thermal hydraulic results are still acceptable at 50° C. Results are presented below for the hottest fuel element.

Parameter	Inlet Temp. 30°C (86°F)	Inlet Temp. 50°C (122°F)
Exit Coolant Temperature, °C (°F)	84.06 (183.3)	98.3 (208.9)
Maximum Wall Temperature, °C (°F)	142.6 (288.6)	142 (288)
Peak Fuel Temperature, °C (°F)	500 (932)	499 (931)
Minimum DNB Ratio	2.50	2.20
Channel Mass Flow Rate, kg/sec	0.0919	0.103
Maximum Flow Velocity, cm/sec	18.90	21.3
Exit Clad Temperature, °C	130.9	131

The results show a reduction in the DNB ratio but very little change in fuel or cladding temperatures. A slight increase in natural circulation flow helps to offset the effect of the higher coolant inlet and exit temperatures.

29. Section 4.8.3. Based on the values for kW/element and rod peaking factor, a simple calculation indicates a MDNBR at 1 MW of 2.50 rather than 2.45 as quoted. Please clarify.

Response:

The hot rod element power of 52 kW is correct. The corresponding reactor power should be 2.50 MW.

30. Section 4.8.3. The SAR states that the location of the highest powered fuel element is D4NE and that this would be an ideal location of the IFE. What consideration has been given to locating the IFE in this location?

Response:

WSU has considered moving an IFE to D4NE.

The replacement core has been predicted to perform substantially similarly to the mixed HEU/LEU core. Placing the IFE's in C4NW and D6NW will allow WSU to generate temperature data on core performance, as a function of power level, for both steady state and pulsing operations. The data for C4NW and D6NW are important because they will be used as bases for comparison to modeling predictions of the LEU core performance. The data will also be useful for benchmarking the new LEU core, which, at this time, has been given the designation of Core 35A.

Shortly after the conversion project is completed, WSU intends to examine other possible core configurations to optimize core characteristics for different kinds of experimental Research and Development work. WSU intends to move to a different core configuration in two steps. The first step would be to move one of the IFE's (either C4NW or D6NW) to D4NE to make an experimental measurement of the peak temperature in D4NE, for comparison with the modeled values. The change would be accomplished by exchanging the fuel bundles in core grid position D4 with one of the bundles equipped with an IFE, i.e. either C4 or D6. The comparison would allow a critical evaluation of the model. The second step would be reconfiguration of the core by changing the physical arrangement of the fuel bundles, reflectors, and irradiation positions. Modeling of candidate core configurations will be carried out before any such changes are made—for that reason it is necessary to develop as much understanding as possible of the accuracy of the modeling for Cores 34A and 35A.

When an acceptable new core configuration is determined, including determination of the location of the highest powered fuel element, it will be necessary to implement a change to the Technical Specifications. At this time the Limiting Safety System Setting (LSSS) for the WSU reactor is 500° C as indicated on the fuel temperature channel. The LSSS value was set on the basis of the temperature ratios that exist between the IFE (in core positions C4NW and D6NW) and the hot rod position at D6NE. Moving an IFE to D6NE for Core 35A would require establishment of two LSSS values, one for D6NE and one for the remaining IFE (either C4NW or D6NW). Additionally, it is not known at this time whether further changes would be required for core configurations subsequent to Core 35A.

31. Section 4.9.1. The description of regions (in cm) of the thermal neutron flux across the reactor core does not appear to match the figures. Please discuss.

Response:

On page 67, section 4.9.1 second paragraph should be:

Figure 38 shows the flux plot through the transient rod in a direction perpendicular to the face of the thermal column/BNCT Filter box. In the region between 25cm and 33 cm the flux through the partially burned 8.5/20 LEU fuel is seen. In the region between 33 cm and 72 cm the flux through the fresh 30/20 LEU fuel (flux depressed) is seen with the water peak in the transient rod. Finally in the region between 72 cm and 80 cm the flux through the opposite region of the partially-burned 8.5/20 LEU fuel is seen.

32. **Section 12.5. Please verify if updates are needed to your physical security plan. If changes are needed and you want them made as part of the conversion process, please submit your updated plan. If changes are needed and you do not want them to be made under conversion, please follow the regulations in 10 CFR 50.54(p). In this case, the changes would need to be in place before the order to convert the reactor is issued.**

Response:

WSU has examined the physical security plan for compliance with 10 CFR 50.54 and is in the process of making necessary modifications for submittal to the U.S. NRC.

33. **Section 13.3. The IFE contains three thermocouples at different locations. What impact will the choice of thermocouple have on the LSSS. This section states that at 1.3 MW, peak fuel temperature in the core is 520 °C. Table 29 indicates a peak fuel temperature of 541 °C. Please explain. The factors listed (items i-iv) to be taken into account when setting the LSSS are termed the "safety margin". However, you discuss a peak core temperature of 950 °C representing a safety margin of 200 °C. Given this, do the factors really represent a safety margin?**

Response:

Of the three thermocouples, the bottom one tends to be the highest due to power tilting toward the lower half of the core from control blade insertion. As an example, for Core 35A at 1 MW, the IFE in D6NW reads 427° C at the bottom, 424° C at the middle, and 414° C at the top thermocouple locations.

The peak fuel temperature at 1.3 MW is 541° C and the text in section 13.3 should be corrected.

The factors do not represent a safety margin and should not be identified as such.

The Limiting Safety System Setting is 500° C for the IFE's (currently in grid positions D6NW and C4NW). It can be shown that the LSSS protects the Safety Limit of 1150° C for the peak temperature on the hot rod, D4NE. The lowest Rod Power Factor for any grid position is 1.22 for Core 35A, BOL at E6SE. The IFE temperature at this position is 408° C at 1.0 MW and 440° C at 1.3 MW. Putting an IFE in this position would give Peak to indicated Temperature Ratio (PTR) values of

$$500° \text{ C}/408° \text{ C} = 1.23$$

$$541° \text{ C}/440° \text{ C} = 1.23$$

If the LSSS for this worst case IFE placement is 500° C, the peak fuel temperature corresponding to an IFE temperature of 500° C would be

$$1.23 \times 500° \text{ C} = 615° \text{ C}$$

Thus, the LSSS value protects the Safety Limit of 1150° C, even if the IFE is placed in the most unfavorable allowed core position. Additionally, the peak fuel temperature at 1.3 MW is 541 ° C, which is less than the 615° C peak fuel temperature that would result from an indicated temperature of 500° C at E6SE; as a result, the high power scram would activate before the LSSS IFE temperature setpoint is reached.

34. Section 13.5.1. Please verify that the references in the text of this section correctly match the reference list in the SAR.

Response:

References 14 and 15 should be replaced with the following:

14. "Amendment 1 to Safety Analysis Report of October, 1966" for the use of FLIP fuel in the WSU reactor, Washington State University, May 1974.

15. "Safety Analysis Report for the Washington State University Modified TRIGA Nuclear Reactor," Washington State University, May 1979.

35. Section 13.5.1. The 2002 SAR has not been approved by NRC. Please submit a stand-alone evaluation for the MHA stating all assumptions and showing calculations. You give occupational doses based on 5-minute and 1-hour stay times. Please base your calculations on a realistic evacuation time based on drill performance.

Response:

The stand-alone analysis is included with this document as Appendix MHA: Analysis of Maximum Hypothetical Accident.

36. Section 13.5.2. Please submit a stand-alone evaluation for the LOCA stating all assumptions and showing calculations. Also, please state what is the margin to clad yield strength with a power density of 22.9 kW/rod.

Response:

A standalone evaluation for the LOCA has been prepared, and is included with this document as Appendix LOCA: Analysis of Loss-of-Coolant Accident (LOCA)

37. Section 13.5.3. Your proposed TS changes discuss limiting the reactivity value of the pulse rod in the pulsing mode by mechanical means or the rod extension physically shortened. Does this impact the accidental pulsing analysis? If so, how?

Response:

The text from this section was inadvertently lifted from the Texas A & M Safety Analysis Report. WSU does not propose technical specification changes for limiting pulse reactivity.

38. Section 14. Provide replacement TS pages with the changes to the TSs shown by change bars in the page margins. Separately, list each change requested to the TS along with a justification for the requested change. In Section 14.1 it appears that you have proposed numerous changes to this section of the TS that are not directly related to conversion. Please ensure that proposed changes to the TSs are related to conversion.

Response:

Proposed Technical Specification changes are attached to this document as Appendix Technical Specifications. The current Technical Specifications are attached to this document as Appendix Technical Specifications Amendment 18. A revised version of the conversion SAR is also included, which contains the updated version of the proposed changes to the Technical Specifications.

39. Section 14.2.1. It appears that you are proposing to change the safety limit for a standard TRIGA fuel rod from 1000 °C to 1150 °C. Please provide a justification for this change given that fuel under the current 1000 °C limit will continue to be used in your reactor.

Response:

WSU is not proposing changes to the Safety Limit for 8.5/20 LEU fuel. The statement in section 14.2.1 was intended to apply only to 30/20 LEU fuel. The Safety Limit for 8.5/20 will remain at 1000° C.

40. Section 14.2.2. Section 13.3 of the SAR evaluates the LSSS for two specific IFE locations. However, the TS allows the IFE to be located anywhere in the 30/20 fuel region. Please repeat the analysis for the worse case to show that the LSSS as proposed protects the safety limit.

Response:

The worst IFE location is E6SE which has an RPF of 1.224 and an APF of 1.47. The rod power is lowest among the 30/20 LEU fuel because it is in the far SE corner. The APF is high because it is next to a control blade. An analysis of this location gives an IFE temperature of 408 °C at 1.0 MW and 440 °C at 1.3 MW. The IFE temperature is 19 °C lower at 1.0 MW and 17°C lower at 1.3 MW than the IFE located at D6NW. This relatively modest reduction in IFE temperatures show that the fuel temperature LSSS along with the power trip at 125% protect the safety limit with margin.

The following text also appears in the Response to Question 33:

The Limiting Safety System Setting is 500° C for the IFE's (currently in grid positions D6NW and C4NW). It can be shown that the LSSS protects the Safety Limit of 1150° C for the peak temperature on the hot rod, D4NE. The lowest Rod Power Factor for any grid position is 1.22 for Core 35A, BOL at E6SE. The IFE temperature at this position is 408° C at 1.0 MW and 440° C at 1.3 MW. Putting an IFE in this position would give Peak to indicated Temperature Ratio (PTR) values of 1.23, as illustrated below:

$$500^\circ\ C/408^\circ\ C = 1.23$$

$$541^\circ\ C/440^\circ\ C = 1.23$$

If the LSSS for this worst case IFE placement is 500° C, the peak fuel temperature corresponding to an IFE temperature of 500° C would be

$$1.23 \times 500^\circ\ C = 615^\circ\ C$$

Thus, the LSSS value protects the Safety Limit of 1150° C, even if the IFE is placed in the most unfavorable allowed core position. Additionally, the peak fuel temperature at 1.3 MW is 541 ° C, which is less than the 615° C peak fuel temperature that would result from an indicated temperature of 500° C at E6SE; as a result, the high power scram would activate before the LSSS IFE temperature setpoint is reached.

41. Section 14.3.1. For TS 3.3, Pulse Mode Operation, it appears that you are proposing to eliminate the requirement to determine the maximum safe allowable reactivity insertion. Please justify elimination of the requirement.

Response:

Initially, the pulse insertions shall be increased by small increments to a maximum of \$2.00 to allow an extrapolation of peak temperatures, thereby establishing the maximum allowed pulse insertion for the LEU core. WSU plans to begin pulse testing with reactivity insertions less than \$1.00 or slightly more than \$1.00 (e.g. \$0.75 and values close to \$1.00, such as \$0.95, \$0.98, \$1.00, \$1.02, and \$1.05) to monitor IFE behavior, reproducibility, and pulse rod calibration. Continued pulse testing will be done by increasing increments of no more than \$0.10 (i.e. \$1.10, \$1.20 etc.) to generate sufficient data to make plots of energy release vs. reactivity and peak power vs. prompt reactivity. Multiple pulses will be conducted for each reactivity value to generate statistical data on reproducibility and pulse rod calibration. The data generated during test pulsing will be examined after each incremental increase in pulse size to determine peak energy density and fuel temperatures (both measured and calculated), for comparison with model predicted values. In no case will the reactor be pulsed greater than \$2.00 unless it can be shown with a high degree of certainty that it can be done without violation of Technical Specification limitations.

42. Section 14.3.1. The TS "Maximum Excess Reactivity" states "The maximum reactivity in excess of cold, xenon-free critical shall not exceed 5.6% Δk/k (\$8.00)". If β=0.0075, then it appears the worth is \$7.47. Please clarify.

Response:

\$8.00 should be replaced with \$7.47

The technical specification that states, "The maximum reactivity in excess of cold, xenon-free critical shall not exceed 5.6% Δk/k (\$8.00). The conversion of 5.6% Δk/k to \$8.00 was done in an earlier SAR, using β_{eff} = 0.007 as the proportionality constant. Refinement of the value of βeff to 0.0075 yields an equivalent excess reactivity of \$7.47 as the reviewer has pointed out. The current best value should be modified to \$7.47.

43. Section 14.3.2. It appears that you are proposing the addition of 2 high power level measuring channels. What is the relationship between the high power level channels and the linear and log power channels? How is this change related to conversion of the reactor?

Response:

A revised version of the conversion SAR, with the updated Technical Specifications is included with this Response to Request for Additional Information.

The referenced table in section 14.3.2 should appear as follows:

Measuring Channel	Min. No. Operable	Effective Mode	
		S.S.	Pulse
Fuel Element Temperature	1	X	X
Linear Power Level	1	X	
Log Power Level	1	X	
Integrated Pulse Power	1		X

44. Section 14.3.2. Why is the manual scram, pool level and interlock that prevents energizing the pulse circuit when the power is less than 1 kW not part of the safety channels that must be operable?

Response:

A revised version of the conversion SAR, with the updated Technical Specifications is included with this Response to Request for Additional Information.

The table in Section 14.3.2 should include the manual scram, pool level, and pulse inhibit circuit. The safety channels are described in the following table.

Minimum Reactor Safety Channels

		Number operable in specified mode	
Safety Channel	Function	SS	Pulse
Fuel temperature	Scram if fuel temperature exceeds 500°C	1	1
Power level	Scram if power level exceeds 125% of full licensed power	1	
Manual scram	Manually initiated scram	1	1
Wide range	Prevent initiation of a pulse above 1 kW		1
	Prevent control element withdrawal when neutron count is less than 2 cps	1	
High-voltage monitor	Scram on loss of high voltage to power channels	1	1
Pulse-mode switch	Prevent withdrawal of standard control and regulation elements in pulse mode		1
Preset timer	Transient rod scram 15 seconds or less after pulse		1
Pool level	Alarm if pool level falls below 16 ft over the core	1	1
Transient rod control	Prevent application of air unless fully inserted	1	

Note: SS = steady-state

45. Section 14.5.2. To the extent that this TS repeats requirements given in the Core Configuration Limitation TS, you may remove the redundancy.

Response:
Ok

46. Appendix A. What action is taken if an acceptance criterion for a startup test is not met?

Response:

The following contingency plans have been added to the startup tests in the event that an acceptance criterion is not met.

A.1.1 Initial Criticality

Acceptance Criteria: The 1/M criticality is expected with a fuel loading as follows:

Partially Burned 8.5/20 SFEs from 24 to 32 rods
Fresh 30/20 SFEs from 43 to 47 rods

Contingency for Failure to Meet Acceptance Criteria

A failure of agreement of data generated by the inverse multiplication/critical loading test with the predicted number of fuel elements would suggest that either the inverse multiplication data are erroneous, or the predictive model is inaccurate. Resolution will be made by reexamining fuel loading procedures, instrument readings, data and calculations, and the predictive model.

A.1.2 Critical Mass and Criticality Conditions for the 30/20 LEU Core

Acceptance Criteria: The Rod Drop reactivity worths for the scrammable control rods are expected to lie between $0.40 and $3.20.

Contingency for Failure to Meet Acceptance Criteria

The reactivity worths for the scrammable control rods will be dependent upon the identity of the control rod, and core configuration at the time that the reactivity worths are determined. It is likely that the worth of control blade number five (the non-scrammable stainless steel regulating blade) will be less than $0.40. If the measured worth of a scrammable control rod or blade falls outside the Acceptance Criteria the deviation will be recorded, the cause determined, and will be reported to the Reactor Safeguards Committee. The WSU Facility Director (a licensed Senior Reactor Operator) and the Reactor Supervisor will consult with General Atomics and the Reactor Safeguards Committee to assess operational and safety implications. The reactor will be shut-down and secured, and use of the reactor, including movement of fuel into or out of the reactor or maintenance of reactor control systems will not be permitted until the WSU Facility Director, Reactor Supervisor, General Atomics, and the Reactor Safeguards Committee agree that continuation of refueling activities may be done safely.

A.1.3 Initial Control Rod Calibration Tests

Acceptance Criteria: The calibration curve results for control rod worth are expected to vary between a low value of approximately $0.20 and a high value of $3.00 to $4.00, depending on the type of rod and location in the core.

Contingency for Failure to Meet Acceptance Criteria

If the measured worth of a control rod or blade falls outside the Acceptance Criteria the deviation will recorded, the cause determined, and will be reported to the Reactor Safeguards Committee. The WSU Facility Director (a licensed Senior Reactor Operator) and Reactor Supervisor will consult with General Atomics and the Reactor Safeguards Committee to assess operational and safety implications. The reactor will be shut-down and secured, and use of the reactor, including movement of fuel into or out of the reactor or maintenance of reactor control systems will not be permitted until the WSU Facility Director, the Reactor Supervisor, General Atomics, and the Reactor Safeguards Committee agree that continuation of refueling activities may be done safely.

A.1.4 Final Core Loading/Final Rod Calibrations

Acceptance Criteria:

- With 119 fuel elements the excess reactivity is expected to be about $ 6.37, the computed value.
- With the same core and same location, the "shut down margin" the shall be greater than $0.25 with the most reactive scrammable rod and the stainless steel regulating blade in the fully withdrawn positions.

Contingency for Failure to Meet Acceptance Criteria

If the excess reactivity value is not within the range $5.87 - $6.87 the deviation will recorded, the cause determined, and will be reported to the Reactor Safeguards Committee. The WSU Facility Director (a licensed Senior Reactor Operator) and Reactor Supervisor will consult with General Atomics and the Reactor Safeguards Committee to assess operational and safety implications. The reactor will be shut-down and secured, and use of the reactor, including movement of fuel into or out of the reactor or maintenance of reactor control systems will not be permitted until the WSU Facility Director, the Reactor Supervisor, General Atomics, and the Reactor Safeguards Committee agree that continuation of refueling activities may be done safely.

The minimum shut down margin shall be $0.25, with the most reactive scrammable control rod and the stainless steel regulating rod fully withdrawn from the core. If this Acceptance Criterion is not met the reactor shall be shut down and secured. The reactor shall not be operated with a shutdown margin that is less than $0.25. If it is determined that the shutdown margin is less than $0.25 the WSU Facility Director and the Reactor Supervisor will consult with General Atomics and the Reactor Safeguards committee to determine an appropriate remedial course of action.

Possible remedial actions include removing fuel or reflector from the reactor until a determination is made of cause and remedy of the smaller than expected shutdown margin.

A.1.5 Calorimetric Reactor Power Calibration

Acceptance Criteria: After the final power calibration, all power channel indications will agree within 2% at full reactor power, 1.0 MW.

Contingency for Failure to Meet Acceptance Criteria

If the power channel indications do not agree within 2% at full reactor power (1 MW) the power calibration will be repeated.

If the power channel indications do not agree within 2% after a second calibration, the WSU Facility Director and the Reactor Supervisor will consult with General Atomics and the Reactor Safeguards Committee to determine an appropriate course of action.

A.1.6 Initial Approach to Full Power

Acceptance Criteria: At full power (1.0 MW), the reactivity loss is expected to lie in the range from \$2.25 to \$2.75, values that verify the presence of a large negative coefficient of reactivity.

Contingency for Failure to Meet Acceptance Criteria

The observed negative temperature coefficient of reactivity is expected to be bracketed by the \$2.25 to \$2.75 range. If the observed value lies outside the Acceptance Criteria range the WSU Facility Director (a licensed Senior Reactor Operator) and the Reactor Supervisor will consult with General Atomics and the Reactor Safeguards Committee to assess operational and safety implications.

Acceptance Criteria: A log-log plot of the detector indications versus the D.C. return current in a fission counter shall be a nearly straight line over a power span from about 100 kW to 1.0 MW, thus demonstrating detector channel linearity.

Contingency for Failure to Meet Acceptance Criteria

Correct functioning of the log power channel is required as one of the minimum number of measuring channels as a Limiting Condition of Operation.

Indications of lack of accurate detector performance shall be investigated and appropriate corrective action taken, e.g. repair or replacement of malfunctioning components. The reactor may only be operated when the log power channel is operating correctly.

Acceptance Criteria: The reactor shall scram reliably when a Safety power channel reaches an indicated 1.25 MW.

Contingency for Failure to Meet Acceptance Criteria

Correct functioning of the high power scram on the Safety Channel is a Limiting Condition of Operation. The reactor shall not be operated unless the high power scram operates correctly.

A.1.7 Pulsing Mode of Operation

Acceptance Criteria: Plots of energy release and peak power for pulsing performance shall be consistent with a linear (straight line) dependence of either (Δk_P) or $(\Delta k_P)^2$, as appropriate.

Contingency for Failure to Meet Acceptance Criteria

Plots of energy release and peak power for pulsing performance will be compared with similar plots prepared from pulsing data previously generated at the WSU reactor. Historical pulsing data will be used to establish estimated uncertainty in measured values for energy release and peak power. The WSU Facility Director and the Reactor Supervisor will consult with General Atomics and the Reactor Safeguards Committee to assess operational and safety implications if plots of energy release and peak power for pulsing performance do not follow the expected linear trends, within experimental uncertainty.

The following text also appears as an answer to Question 41:

Initially, the pulse insertions shall be increased by small increments to a maximum of \$2.00 to allow an extrapolation of peak temperatures, thereby establishing the maximum allowed pulse insertion for the LEU core. WSU plans to begin pulse testing with reactivity insertions less than \$1.00 or slightly more than \$1.00 (e.g. \$0.75 and values close to \$1.00, such as \$0.95, \$0.98, \$1.00, \$1.02, and \$1.05) to monitor IFE behavior, reproducibility, and pulse rod calibration. Continued pulse testing will be done by increasing increments of no more than \$0.10 (i.e. \$1.10, \$1.20 etc.) to generate sufficient data to make plots of energy release vs. reactivity and peak power vs. prompt reactivity. Multiple pulses will be conducted for each reactivity value to generate statistical data on reproducibility and pulse rod calibration. The data generated during test pulsing will be examined after each incremental increase in pulse size to determine peak energy density and fuel temperatures (both measured and calculated), for comparison with model predicted values. In no case will the reactor be pulsed greater than \$2.00 unless it can be shown with a high degree of certainty that it can be done without violation of Technical Specification limitations.

47. Appendix A.1.1. It is stated that criticality is expected with a loading of 58-68 fresh, 30/20 fuel elements. The acceptance criteria states that 1/M criticality is expected with 43-77 30/20 fuel elements and 24-32 partially burned standard fuel elements. Please explain the difference in these two statements.

Response:

In the question, 43-77 30/20 fuel elements should be 43-47 30/20 fuel elements (refer to page 96 of the SAR).

The first statement indicates that, based on experience with other TRIGA reactors, the number of fresh 30/20 fuel rods needed to reach cold critical is about 58 to 68 without any 8.5/20 and FLIP fuel elements. However, the WSU reactor will be a mixed core with 8.5/20 and 30/20 fuel. Core 35A will be composed of twelve 4-rod 30/20 fuel assemblies, one 3-rod 30/20 fuel assembly, and seventeen 4-rod 8.5/20 fuel assemblies. Thus, there will only be fifty-one 30/20 fuel rods in Core 35A, which are fewer than predicted to be required to achieve criticality. As a result, the fuel loading plan will require the use of both 8.5/20 and 30/20 fuels. Consequently, the second statement is adopted as the Acceptance Criteria for Initial Criticality, i.e.

The 1/M criticality is expected with a fuel loading as follows:

Partially Burned 8.5/20 SFEs from 24 to 32.
Fresh 30/20 SFEs from 43 to 47.

48. Appendix A.1.4. The SAR implies that the regulating rod does not have scram capability. However, the acceptance criterion for shutdown margin only refers to the most reactive rod stuck out of the core. Please clarify.

Response:

The regulating control blade (number 5) does not have scram capability. Minimum shutdown margin is determined with the most reactive control blade and the regulating rod completely withdrawn. The second acceptance criterion in Appendix A.1.4 should state:

- With the same core and same location, the "shutdown margin" with the most reactive rod and regulating rod (number 5) fully withdrawn from the core will be greater thand $0.25

Appendix A
Fuel Engineering Diagrams

The engineering diagrams for standard TRIGA fuel and the Instrumented Fuel Element are presented on the following pages.

Appendix MHA

Analysis of Maximum Hypothetical Accident

The Maximum Hypothetical Accident (MHA) for a TRIGA reactor is defined as the loss of the integrity of the fuel cladding of one fuel rod in air. This MHA definition is consistent with the Design Basis Accident (DBA) defined in NUREG/CR-2387 (Ref. 13).

To compare the relative abundance of fission products, the pertinent operating parameters are compared for a FLIP core and a LEU (30/20) core. The major parameter important to radiological impact is the fission product inventory due to burnup and power density. The WSU mixed HEU Core 34A has a life of about 1000 MWD and is essentially the same as the WSU mixed LEU Core 35A as discussed in Section 4.5.6. The energy burnup capability for FLIP fuel is 77 MWD which is much greater than 50% burnup. For WSU LEU (30/20) fuel, the energy burnup capability based on 50% of the U-235 gives 54 MWD/fuel element (NUREG-1282).

On a per MWD basis, the fission product inventory for TRIGA HEU fuel and LEU fuel are negligibly different. For TRIGA HEU fuel, less than 1% of the fission power at EOL has come from Pu-239. However, for TRIGA LEU fuel, the percentage of fission power from Pu-239 increases to less than 4%. In both fuels, almost all of the fission product inventory is due to U-235 fission. Any inventory difference between HEU and LEU fuel for a given burnup is minor compared to other uncertainties in the assessment of radiological dose.

The fission product inventories in Table 30 were calculated using ORIGEN. Both FLIP and LEU (30/20) inventories were calculated for a single fuel element. The element was assumed to have a conservative power density of 28 kW. The burnup calculation was performed for 200 days (5.6 MWD) in order to achieve saturation levels for all of the isotopes except Kr-85. Maximum inventories occurred after 200 MWD of burnup which maximized buildup while minimizing the effect of fissile material depletion. The burnup calculation for Kr-85 was extended to 77 MWD for FLIP fuel and 54 MWD for LEU (30/20) fuel.

The rate at which fission products are removed from the pool room and released into the environment during an MHA depends on the removal rate of pool room air by the ventilation system. In the normal operation mode, air is exhausted from the pool room at the rate of 4500 cfm (2.12×10^6 cm^3/sec). On detection of high radiation levels by the Continuous Air Monitor (CAM), the HVAC switches to dilution mode. In the dilution mode, 300 cfm (1.41×10^5 cm^3/sec) of air from the pool room is passed through a HEPA filter system, diluted with 1700 cfm of outside air and discharged to the atmosphere. In the dilution mode, 2000 cfm (9.44×10^5 cm^3/sec) of air is discharged with a dilution factor of 6.67. In the dilution mode, the HEPA filter would remove at least 90% of the iodine and particulates from the exhaust air. If the ventilation system is switched to isolation mode, the release to the environment would only be by leakage from a sealed building which is estimated to be of the order of 100 cfm (4.72×10^4 cm^3/sec). The analysis of the MHA assumes that the ventilation is automatically switched to the dilution mode but that the HEPA filter is ineffective in removing volatile fission products from the exhaust air.

Table 30
Gaseous Fission Products in a Single TRIGA Fuel Rod (28 kW/rod)

Isotope	FLIP (Curies)	LEU (30/20) (Curies)	Half-Life
Br-82	1	1	35.3 hr
Br-83	127	126	2.3 hr
Br-84	246	245	31.8 min
Br-85	291	288	3.0 min
I-131	681	685	8.1 days
I-132	1022	1029	2.3 hr
I-133	1587	1590	21.0 hr
I-134	1829	1829	54.0 min
I-135	1488	1488	6.8 hr
Kr-83m	127	126	1.9 hr
Kr-85m	290	288	4.4 hr
Kr-85	23	18	10.7 yr
Kr-87	594	589	78.0 min
Kr-88	840	834	2.8 hr
Kr-89	1095	1084	3.2 min
Xe-131m	7.5	7.5	12.0 days
Xe-133m	46	46	2.3 days
Xe-133	1547	1550	5.3 days
Xe-135m	275	276	15.0 min
Xe-135	962	1057	9.0 hr
Xe-137	1449	1449	3.9 min
Xe-138	1504	1504	17.0 min

A fuel element with a conservative maximum temperature of 535° C has an average fission product release fraction of 2.6 x 10^{-5} into the clad gap. This release fraction is averaged over the fuel volume and temperature. Evacuation drills have demonstrated that personnel within the pool room can be evacuated within 5 minutes. The 5 minute occupational exposure during an MHA in which a single LEU 30/20 fuel rod fails in air is affected by radioactive decay and pool room ventilation during the 5 minute exposure. The time-integrated concentration during the 5-minute exposure is determined by the following equation:

$$\overline{C}(0 - 300\,s) = \frac{A}{V(\lambda_L + \lambda_D)}\left[1 - e^{-(\lambda_L + \lambda_D)300\,s}\right]$$

where A = activity released into pool room,

V = pool room volume,
λ_L = pool room ventilation rate = 1.41×10^{-4} sec^{-1},
λ_D = radioactive decay rate, sec^{-1}.

The fission product release from a single LEU 30/20 fuel element into the pool room atmosphere, resulting time-averaged concentration, and whole body dose are presented in Table 31. The release is instantaneous into the pool room which has a volume of 1×10^9 cm^3. The thyroid dose is presented in Table 32. The whole body and thyroid dose rates are calculated using dose conversion factors (DCFs) from EPA-400-R-92-001, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents. These DCFs include both inhalation and external exposure effects. The 21.5 mrem whole body and 2.77 rem thyroid which are much less than the regulatory limits stated in NUREG-1537 of 5 rem whole body and 30 rem thyroid for research reactors licensed before January 1, 1994.

Table 31
Pool Room Fission Product Concentrations and Associated 5-Minute Whole Body Worker Dose for Single LEU 30/20 Fuel Rod Cladding Failure in Air

Isotope	Release to Pool Room (mCi)	Time-Integrated Concentration (μCi-s/cm^3)	DCF in rem per μCi/cm^3/hr	Whole Body Dose (mrem)
Br-82	0.026	7.63×10^{-6}	1250	2.6×10^{-3}
Br-83	3.3	9.50×10^{-4}	83	0.02
Br-84	6.4	1.77×10^{-3}	125	0.06
Br-85	7.5	1.31×10^{-3}	115	0.04
I-131	17.8	5.23×10^{-3}	220	0.32
I-132	26.8	7.76×10^{-3}	1400	3.02
I-133	41.3	1.21×10^{-2}	350	1.18
I-134	47.6	1.35×10^{-2}	1600	6.01
I-135	38.7	1.13×10^{-2}	950	2.99
Kr-83m	3.3	9.48×10^{-4}	100	0.03
Kr-85m	7.5	2.19×10^{-3}	93	0.06
Kr-85	0.5	1.37×10^{-4}	1.3	5.0×10^{-5}
Kr-87	15.3	4.40×10^{-3}	510	0.62
Kr-88	21.7	6.30×10^{-3}	1300	2.28
Kr-89	28.2	5.08×10^{-3}	1200	1.69
Xe-131m	0.2	5.73×10^{-5}	4.9	7.8×10^{-5}
Xe-133m	1.2	3.51×10^{-4}	17	1.7×10^{-3}
Xe-133	40.3	1.18×10^{-2}	140	0.46
Xe-135m	7.2	1.88×10^{-3}	250	0.13
Xe-135	27.5	8.05×10^{-3}	140	0.31
Xe-137	37.7	7.35×10^{-3}	110	0.22
Xe-138	39.1	1.04×10^{-2}	710	2.05
Total Whole Body Dose				21.50

Table 32
Pool Room Fission Product Concentrations and Associated 5-Minute Thyroid Worker Dose for Single LEU 30/20 Fuel Rod Cladding Failure in Air

Isotope	Release to Pool Room (mCi)	Time-Integrated Concentration (μCi-s/cm^3)	DCF in rem per μCi/cm^3/hr	Thyroid Dose (rem)
I-131	17.8	5.23×10^{-3}	1.3×10^{6}	1.89
I-132	26.8	7.76×10^{-3}	7.7×10^{3}	0.02
I-133	41.3	1.21×10^{-2}	2.2×10^{5}	0.74
I-134	47.6	1.35×10^{-2}	1.3×10^{3}	4.9×10^{-3}
I-135	38.7	1.13×10^{-2}	3.8×10^{4}	0.12
Total Thyroid Dose				2.77

The gaseous radioactive material discharged from the facility ventilation system will be diluted by atmospheric air in the lee of the building due to turbulent wake effects. The nearest member of the public could be as close as 50 ft (15 m) which would place the individual within the 20-m wake cavity formed by the building. Using a Gaussian plume dispersion model, the initial dimensions of the plume are defined by the following parameters (NUREG/CR-4691, 1990):

$$\sigma_y = W_b/4.3$$

$$\sigma_z = H_b/2.15$$

where W_b and H_b are the width and height of the building.

Using a conservative windspeed, u, of 1 m/sec (2.2 mph), a building height of 8.53 m (28 ft) and a minimum building width of 17.07 m (56 ft), the χ/Q for the nearest member of the public is calculated as follows:

$$\frac{\chi}{Q} = \frac{1}{\pi \sigma_y \sigma_z u} = 0.0202 \ s/m^3$$

The nearest member of the public is assumed to remain throughout the accidental release. The fraction of the activity released to the pool room that is discharged to the environment depends on the radioactive decay rate and the pool room exhaust rate while in dilution mode. No credit is taken for deposition mechanisms either within the pool room or in the environment. The fraction of activity discharged to the environment is calculated by the following relationship:

$$\frac{\lambda_L}{\lambda_L + \lambda_D}$$

The ventilation system will have switched to dilution mode during the MHA but no credit is taken for the HEPA filter in reducing iodine release. Therefore the fractional discharge rate from the pool room is 1.41×10^{-4} per second or 8.46×10^{-3} per minute. The resulting environmental release, time-integrated concentration and dose to the whole body and thyroid are presented in Tables 33 and 34. These doses are much less than the 0.5 rem whole body and 3 rem thyroid doses acceptable for members of the public for research reactors licensed before January 1, 1994.

Table 33
Environmental Fission Product Release and Whole Body Exposure
for Single LEU 30/20 Fuel Rod Failure in Air

Isotope	Release to Pool Room (mCi)	Release to Environment (mCi)	Time Integrated Concentration (μCi-s/m^3)	DCF in rem per μCi/cm^3/hr	W.B. Dose (mrem)
Br-82	0.026	0.025	0.51	1250	1.8×10^{-4}
Br-83	3.3	2.06	41.52	83	9.6×10^{-4}
Br-84	6.4	1.78	35.98	125	0.001
Br-85	7.5	0.26	5.34	115	1.7×10^{-4}
I-131	17.8	17.69	357.25	220	0.022
I-132	26.8	16.79	339.10	1400	0.132
I-133	41.3	38.82	784.08	350	0.076
I-134	47.6	18.89	381.60	1600	0.170
I-135	38.7	32.22	650.81	950	0.172
Kr-83m	3.3	1.91	38.50	100	0.001
Kr-85m	7.5	5.72	115.43	93	0.003
Kr-85	0.5	0.47	9.45	1.3	3.4×10^{-6}
Kr-87	15.3	7.47	150.87	510	0.021
Kr-88	21.7	14.58	294.43	1300	0.106
Kr-89	28.2	1.06	21.40	1200	0.007
Xe-131m	0.2	0.19	3.92	4.9	5.3×10^{-6}
Xe-133m	1.2	1.17	23.58	17	1.1×10^{-4}
Xe-133	40.3	39.87	805.41	140	0.031
Xe-135m	7.2	1.11	22.43	250	0.002
Xe-135	27.5	23.86	482.00	140	0.019
Xe-137	37.7	1.71	34.58	110	0.001
Xe-138	39.1	6.72	135.73	710	0.027
Total Whole Body Dose					0.792

Table 34
Environmental Fission Product Release and Thyroid Exposure
for Single LEU 30/20 Fuel Rod Failure in Air

Isotope	Release to Pool Room (mCi)	Release to Environment (mCi)	Time Integrated Concentration (μCi-s/m^3)	DCF in rem per μCi/cm^3/hr	Thyroid Dose (mrem)
I-131	17.8	17.69	357.25	1.3×10^{6}	129.01
I-132	26.8	16.79	339.10	7.7×10^{3}	0.73
I-133	41.3	38.82	784.08	2.2×10^{5}	47.92
I-134	47.6	18.89	381.60	1.3×10^{3}	0.14
I-135	38.7	32.22	650.81	3.8×10^{4}	6.87
Total Thyroid Dose					184.66

The nearest resident to the WSU reactor is at the Valley Crest Village apartments located approximately 600 m (2000 ft) southwest of the facility. Conservative weather conditions of Stability Class F and windspeed of 1 m/s (2.2 mph) are assumed to persist throughout the release from the MHA. Deposition processes and plume meander are neglected. The environmental release is the same for the nearest resident as for the nearest member of the public given in Tables 33 and 34. The atmospheric dispersion parameters, σ_y and σ_z, based on the methodology of NUREG/CR-1641 and DOE/TIC-11223 at 600 m are 27.1 and 11.0 m, respectively. The resulting χ/Q for the nearest resident is 1.07×10^{-3} s/m^3.. The environmental release, time-integrated concentration and dose to the whole body and thyroid are presented in Tables 35 and 36. These doses are much less than the 0.5 rem whole body and 3 rem thyroid doses acceptable for members of the public for research reactors licensed before January 1, 1994.

Table 35
Environmental Fission Product Release and Whole Body Exposure
for Single LEU 30/20 Fuel Rod Failure in Air

Isotope	Release to Environment (mCi)	Time Integrated Concentration (μCi-s/m^3)	DCF in rem per μCi/cm^3/hr	Whole Body Dose (mrem)
Br-82	0.025	0.03	1250	9.3×10^{-6}
Br-83	2.06	2.20	83	5.1×10^{-5}
Br-84	1.78	1.91	125	6.6×10^{-5}
Br-85	0.26	0.28	115	9.0×10^{-6}
I-131	17.69	18.92	220	0.001
I-132	16.79	17.96	1400	0.007
I-133	38.82	41.53	350	0.004
I-134	18.89	20.21	1600	0.009
I-135	32.22	34.47	950	0.009
Kr-83m	1.91	2.04	100	5.7×10^{-5}
Kr-85m	5.72	6.11	93	1.6×10^{-4}
Kr-85	0.47	0.50	1.3	1.8×10^{-7}
Kr-87	7.47	7.99	510	0.001
Kr-88	14.58	15.60	1300	0.006
Kr-89	1.06	1.13	1200	3.8×10^{-4}
Xe-131m	0.19	0.21	4.9	2.8×10^{-7}
Xe-133m	1.17	1.25	17	5.9×10^{-6}
Xe-133	39.87	42.66	140	0.002
Xe-135m	1.11	1.19	250	8.3×10^{-5}
Xe-135	23.86	25.53	140	9.9×10^{-4}
Xe-137	1.71	1.83	110	0.006
Xe-138	6.72	7.19	710	0.001
Total Whole Body Dose				0.042

Table 36
Environmental Fission Product Release and Thyroid Exposure
for Single LEU 30/20 Fuel Rod Failure in Air

Isotope	Release to Environment (mCi)	Time Integrated Concentration (μCi-s/m^3)	DCF in rem per μCi/cm^3/hr	Thyroid Dose (mrem)
I-131	17.69	18.92	1.3×10^6	6.83
I-132	16.79	17.96	7.7×10^3	0.04
I-133	38.82	41.53	2.2×10^5	2.54
I-134	18.89	20.21	1.3×10^3	0.01
I-135	32.22	34.47	3.8×10^4	0.36
Total Thyroid Dose				9.78

"MELCOR Accident Consequence Code System (MACCS): Model Description," NUREG/CR-1691, Vol. 2, February 1990.

Hanna, Steven R., Gary A. Briggs, and Rayford P. Hosker, Jr., "Handbook on Atmospheric Dispersion," DOE/TIC-11223, 1982.

Appendix LOCA

Analysis of Loss-of-Coolant Accident (LOCA)

The strength of the fuel element clad is a function of its temperature. The stress imposed on the clad is a function of the fuel temperature as well as the hydrogen-to-zirconium ratio, the fuel burnup, and the free gas volume within the element. The analysis of the stress imposed on the clad and strength of the clad uses the following assumptions:

1) The fuel and clad are at the same temperature.
2) The hydrogen-to-zirconium ratio is 1.7 for standard fuel (8.5 wt%) and 1.6 for FLIP fuel and 30/20 fuel.
3) A space one-eighth inch high within the clad represents the free volume within the element.
4) The reactor contains fuel that has undergone burnup equivalent to 77 MW-days for FLIP fuel and 54 MW-days for LEU 30/20 fuel.
5) Maximum operating temperature of the fuel is 600° C.

The fuel element internal pressure P is given by:

$$P = P_h + P_{fp} + P_{air}$$

where:

P_h is the hydrogen pressure;

P_{fp} is the pressure exerted by volatile fission products; and

P_{air} is the pressure exerted by trapped air.

For hydrogen-to-zirconium ratios greater than about 1.58, the equilibrium hydrogen pressure can be approximated by:

$$P_h = \exp\left(1.76 + 10.3014x - \frac{19740.37}{T_k}\right)$$

where:

x is the ratio of hydrogen atoms to zirconium atoms, and
T_k is the fuel temperature (K).

The pressure exerted by the fission product gases is given by:

$$P_{fp} = f\frac{n}{E}\frac{RT_k}{V}E$$

where:

f is the fission product release fraction;

$\frac{n}{E}$ is the number of moles of gas evolved per unit of energy produced (mol/MW-day);

R is the gas constant (8.206×10^{-2} L-atm/mol-K);

V is the free volume occupied by the gasses (L); and

E is the total energy produced in the element (MW-day).

The fission product release fraction is given by:

$$f = 1.5\times10^{-5} + 3.6\times10^{3}\exp\left(\frac{-1.34\times10^{4}}{T_o}\right)$$

where:

T_o is the maximum fuel temperature in the element during normal operation (K).

The fission product gas production rate, $\frac{n}{E}$, varies slightly with the power density. The value 1.19×10^{-3} mol/MW-day is accurate to within a few percent over the range from a few kilowatts per element to well over 40 kW per element. The free volume occupied by the gases is assumed to be a space one-eighth inch (0.3175 cm) high at the top of the fuel so that

$$V = 0.3175\pi \cdot r_i^2$$

where:

r_i is the inside radius of the clad (1.745 cm).

For standard TRIGA fuel, the maximum burnup is about 4.5 MW-days per element, but the TRIGA-FLIP fuel is capable of burnup to about 77 MW-days per element. The LEU 30/20 fuel has been tested to 50% burnup so its capability is slightly less than FLIP fuel at 54 MW-days per element. As the fission product gas pressure is proportional to the energy released, assume that the FLIP fuel in the reactor has undergone maximum burnup.

Finally, the air trapped within the fuel element clad will exert a pressure

$$P_{air} = \frac{RT_k}{24}$$

where it is assumed that the initial specific volume of the air is 22.4 L/mol. Actually, the air forms oxides and nitrides with the zirconium, so that after relatively short operation the air is no longer present in the free volume inside the fuel element clad. The results of the stress imposed on the clad for standard and FLIP fuels are in Figure . The stress imposed by LEU 30/20 fuel would be slightly less than the FLIP fuel due to its lower maximum burnup capability. These results confirm the conclusion of NUREG-1282 that the LEU 30/20 fuel has a safety limit of 950 °C when the clad temperature equals the fuel temperature.

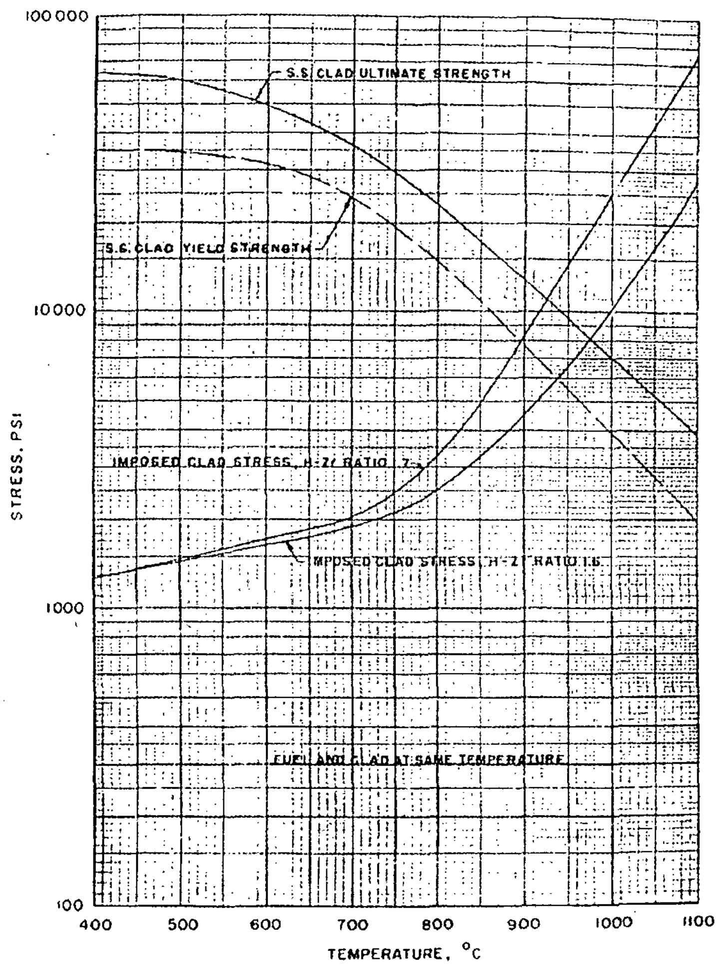


Figure 40: Strength and Applied Stress as a Function of Temperature for 1.7 and 1.6 H-Zr TRIGA Fuel

The maximum fuel cladding temperature after a loss of pool water depends on the fuel rod power density and the time delay between reactor shutdown and uncovering of the core. For the case of no delay, a value of 21 kW/element is reported to prevent fuel temperatures exceeding 900 °C (Foushee, 1972). The analysis developed a two-dimensional transient-heat transport computer code model (TAC2D) for calculating the maximum fuel temperatures after the loss of pool water for various delay times. During the loss of pool water, the low water level alarm occurs when the water level is 18 ft above the top of the core. The time between the actuation of the pool level alarm and the uncovering of the fuel for a catastrophic failure of an eight-inch beam tube is assumed to be 15 minutes (900 sec). Figure 41 presents a curve of maximum fuel cladding temperature versus fuel rod power density using a 15-minute delay. The results show that standard (8.5%) fuel can remain below 900° C at a power density of 22.3 kW/element. The results also show that a FLIP or LEU (30/20) fuel element can remain below 950° C at a power density of 23.5 kW/element. This power density is below the maximum power density of 20.8 kW/element for the WSU reactor operating at 1 MW. Additional delay time is likely since the most likely initiator of a beam tube rupture is the dropping of a large heavy object from above the reactor pool. Such activity is not anticipated for several hours after reactor shutdown.

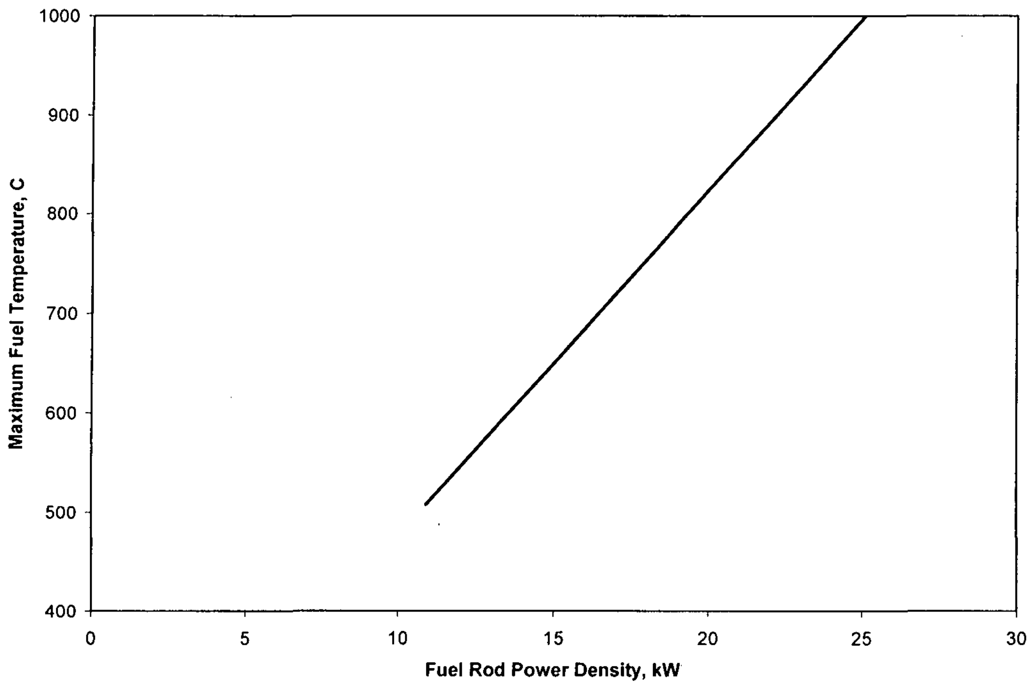


Figure 41: Maximum Fuel Rod Temperature for Loss of Coolant 15 Minutes After Shutdown

If the reactor operates for seventy MW-hours or less per week, power generation per element values approximately 20% higher are sufficient to meet the safety limits. Thus, 26.7 kW/element for standard and 28.2 kW/element for FLIP and LEU 30/20 fuel are adequate power densities. A comparison of decay heat generation versus time following loss of coolant for infinite reactor operations and 70 MW-hours per week cycle operation are in Figure .

Even though the probability of a loss-of-coolant accident is extremely remote, calculations have been performed to evaluate the radiological hazards. The radiation dose rates are given in Table X1 and are based on the assumption that the reactor has been operating for a very long time at a power level of 1 MW prior to the loss of pool water. The times listed in the table are after shutdown of the reactor from full power. The first location is directly on top of the reactor at the bridge level which is 23.5 feet above the top of the actual fueled portion of the core. The second location is at the pool room floor level at the freight door at the east end of the pool room. This second location is shielded from direct radiation from the core but subjected to scattered radiation from the ceiling of the pool room. The ceiling is assumed to be thick concrete yielding the maximum possible reflected radiation dose which is a conservative assumption since the actual roof structure would yield much less scattered radiation.

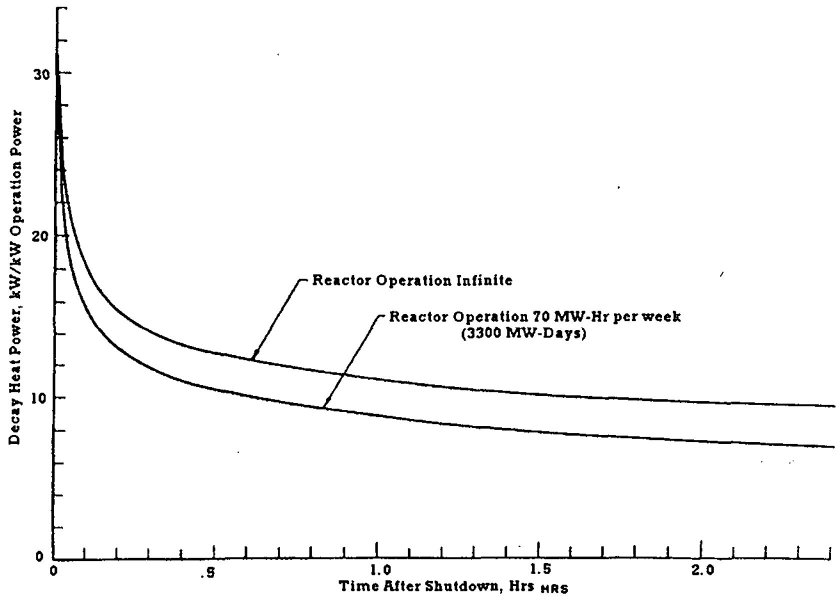


Figure 42: Decay Heat Power Generation Following Loss of Coolant for Infinite Reactor Operation and Periodic Reactor Operation

Table X1
Calculated Radiation Exposure Rates in a Loss-of-Pool-Water Accident

Time After Shutdown	Reduction Factor Due to Decay	Direct Radiation (rem/hr)	Scattered Radiation (rem/hr)
10 sec	1	7350	0.25
1 hr	2.70	2720	0.093
1 day	8.67	848	0.025
1 week	18.84	396	0.015
1 month	72.20	102	0.0035

The data given in Table X1 was calculated assuming that the bare unshielded core is a cylindrical source of 1 MeV photons with a uniform source distribution. The dimensions of the cylinder were taken equal to the active core lattice which has an equivalent radius of 29.1 cm, height of 38.1 cm, and a volume of 5.75×10^4 cm^3. The source strength as a function of time was determined from Perkins and King's data on fission product decay. No accounting was made for sources other than fission product decay gammas or for attenuation through the fuel rod end pieces, core support structure, or bridge deck plate. It is also assumed that no buildup occurs in the core. The sum total effect of these assumptions is a conservative (over estimation) of the dose rates.

Foushee, F. C., "TRIGA Four-Rod Cluster Loss-of-Coolant Accident Analysis," Gulf General Atomics, E-117-196, October, 1972.

Perkins, J. F., and King, R. W., "Energy Release from the Decay of Fission Products," Nuclear Science and Engineering, vol. 3, p. 726, 1968.